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Volume 1 of 5

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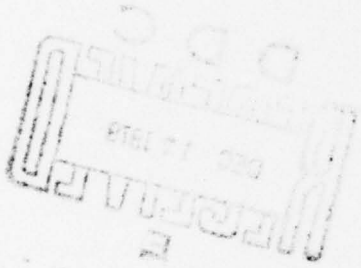

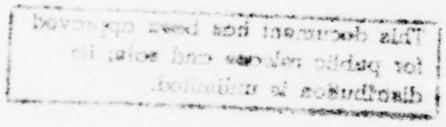

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1.0 SUMMARY

1.1 BASES AND BACKGROUND INFORMATION

ABSTRACT

1.0 SUMMARY

This report was prepared in support of the "Environmental Impact Statement on Management of Commercially Generated Radioactive Wastes", DOE/EIS-0046-D. The scope of this report is limited to technology for management of post-fission wastes produced in the commercial nuclear power light-water reactor (LWR) fuel cycle. Management of spent fuel (as a waste), high-level and other transuranic (TRU) wastes, and gaseous wastes are characterized. Non-transuranic wastes are described but management of these wastes, except for gaseous wastes, is excluded from the scope of this report.

ABSTRACT

Information was developed in sufficient detail to enable environmental impact assessments of the waste management activities to be made. These assessments are contained in a companion document, "Environmental Aspects of Commercial Radioactive Waste Management," DOE/ET-0029.

The contents of this report and the location where each topic is discussed in this summary are as follows:

Report Section	Title	Summary Section
3	Bases and Background Information	1.1
4	Waste Treatment Technology	1.2
5	Interim Storage Technology	1.3
6	Waste Transportation Technology	1.4
7	Final Isolation Technology	1.5
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1.1 BASES AND BACKGROUND INFORMATION

Information contained in the section on bases and background information includes:

Report Section	Section 3 Topics	Summary Section
3.1	Fuel Cycle Options	1.1.1
3.2	Primary Fuel Cycle Facilities	1.1.2
3.3	Waste Descriptions and Classifications	1.1.3
3.4	Waste Management Alternatives	1.1.4
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1.1.1 Fuel Cycle Options

The three basic nuclear fuel cycle options considered in this document are: 1) the once-through cycle, 2) the uranium-only recycle case, and 3) the uranium-plutonium recycle case. Two variations are also considered: 1) the deferred cycle, where the decision for disposal or reprocessing of spent fuel is delayed for a number of years, and 2) delayed repository availability.

In the once-through fuel cycle, irradiated fuel assemblies are isolated as a waste without reprocessing. The reference once-through cycle provides 6.5 years storage of unpackaged spent fuel in water-cooled storage basins (either at the nuclear power plant or at an offsite independent spent fuel storage facility), followed by packaging and final isolation in a geologic repository.

The underlying assumption of the uranium-only recycle case is that spent fuel will be reprocessed to recover the residual uranium for recycle without plutonium recycle. In the reference uranium-only recycle case it is also assumed that all spent fuel storage requirements can be met by nuclear power plant storage basins and reprocessing plant basins. During reprocessing, uranium and plutonium are separated and purified by the Purex solvent extraction process; the uranium is converted to UF_6 and returned to the enrichment plant and plutonium is either 1) stored in an oxide form for future use or disposal, or 2) combined with the high-level liquid waste during solidification and ultimately sent to a geologic repository.

In the plutonium and uranium recycle case, the purified plutonium product is converted to PuO_2 and then blended with UO_2 for incoporation in mixed-oxide (MOX) fuel elements for recycle. The recycle uranium is converted to UF_6 and returned to a uranium enrichment plant as in the uranium-only recycle case.

1.1.2 Primary Fuel Cycle Facilities

The primary fuel cycle facilities in which the post-fission fuel cycle wastes originate include:

1. Nuclear power plant. A 1200 MWe plant is described.
2. Independent spent fuel storage basin (ISFSB). A basin having capacity to store fuel assemblies containing 3000 MTHM and to receive and ship such elements at rates of 500 MTHM/yr is described.
3. Fuel reprocessing plant (FRP). A 2000 MTHM/yr plant is described.
4. Mixed-oxide fuel fabrication plant (MOX FFP). A 400 MTHM/yr plant is described.

These facility descriptions provide a basis for estimating the quantities and characteristics of the wastes to be treated and provide perspective for waste management costs and construction requirements.

1.1.3 Waste Descriptions and Classifications

The primary wastes, which result from operation of the primary facilities, are classified by radiation level and by transuranic element content as well as by waste category (e.g., gaseous, liquid, combustible solid, failed equipment, etc.). Both the non-TRU and TRU wastes from the post-fission LWR fuel cycle are described even though non-TRU waste management, except for post-fission gaseous wastes, is outside the scope of this report.

Data are summarized in Tables 1.1.1 and 1.1.2 for the primary wastes for both the once-through and the plutonium plus uranium recycle cases. Wastes for the uranium-only recycle case are in most cases not markedly different from those of the plutonium plus uranium recycle case.

TABLE 1.1.1. Primary Wastes from Facilities Generating TRU Wastes

Facility and Waste Type	Fuel Cycle	Volume, m^3/MTHM	Radionuclide Content, Ci/MTHM[a]		
			Fission Products	Actinides	Activation Products
Nuclear power plant	Once-Through				
Spent fuel[b]		0.4	3×10^6	1×10^5	2×10^4
FRP[c]	Recycle				
Fuel residue		0.32	8×10^2	1×10^2	9×10^3
High-level liquid waste		0.6	1×10^6	2×10^4	--
Gaseous wastes		1.8×10^6	8×10^3	4×10^{-2}	6×10^{-1}
Combustible and compactable wastes		1.8	2×10^1	1×10^2	--
Miscellaneous liquid and particulate solid wastes		0.15	2×10^2	2×10^2	--
Failed equipment and noncombustible wastes		0.65	3	4×10^1	--
MOX FFP	Recycle				
Gaseous wastes		2.2×10^5[d]	--	4×10^{-2}	--
Combustible and compactable waste		0.12[d]	--	1×10^2	--
Miscellaneous liquid and particulate solid wastes		0.074[d]	--	2×10^1	--
Failed equipment and noncombustible wastes		0.080[d]	--	1×10^1	--

a. Ci/MTHM of spent fuel; in the MOX FFP cases, multiply by 5 to obtain Ci/MTHM fabricated. Based on 0.5 yr after discharge for spent fuel, 1.5 yr after discharge for FRP, and 1 yr after reprocessing for MOX FFP.
b. Spent fuel is a waste only in the once-through fuel cycle.
c. Excluding some wastes from the storage basin and UF_6 conversion portions of the FRP (which are included in Table 1.1.2).
d. m^3/MTHM reprocessed. Multiply by 5 to obtain m^3/MTHM of MOX fuel fabricated.

TABLE 1.1.2. Primary Wastes from Facilities Generating Non-TRU Wastes

Facility and Waste Type	Fuel Cycle	Volume, m^3/MTHM	Radionuclide Content, Ci/MTHM[a]		
			Fission Products	Actinides	Activation Products
Nuclear power plant	Recycle or once-through				
Combustible and compactable wastes		5.4	2×10^{-1}	--	2×10^{-1}
Miscellaneous liquid and particulate solid wastes		9.5	1×10^{2}	--	4×10^{1}
Failed equipment and noncombustible wastes		1.2	5×10^{-2}	--	5×10^{3}
ISFSB	Once-through				
Gaseous wastes		7×10^{6}	8×10^{-2}	--	4×10^{-6}
Combustible and compactable wastes		1.2	2×10^{-2}	--	4×10^{-3}
Miscellaneous liquid and particulate solid wastes		0.052	3	--	6×10^{-1}
Failed equipment and noncombustible wastes		0.16	2×10^{-3}	--	4×10^{-4}
FRP[b]	Recycle				
Gaseous wastes		1.3×10^{6}	8×10^{-7}	5×10^{-5}	--
Combustible and compactable wastes		0.66	3×10^{-2}	3×10^{-3}	6×10^{-3}
Miscellaneous liquid and particulate solid wastes		0.59	2	7×10^{-3}	5×10^{-1}
Failed equipment and noncombustible wastes		0.06	1×10^{-3}	--	3×10^{-4}

a. Based on 0.5 yr after discharge for nuclear power plant, 3.5 yr after discharge for ISFSB, and 1.5 yr after discharge for FRP.
b. Some wastes from the storage basin and UF_6 conversion portions of the FRP.

1.1.4 Waste Management Alternatives

Alternative methods described for managing the various TRU and gaseous wastes are summarized in Table 1.1.3. In many cases, other alternatives were also considered, but are not described in detail. Many of the interim storage alternatives were evaluated for independent federal locations as well as for on-site application.

Final dispositon of treated wastes will involve isolation in a geologic medium; salt is the reference medium but others were considered and described. The waste management alternatives of Table 1.1.3 that are indicated by asterisks make up the remainder of the reference integrated waste management system for the indicated fuel cycle.

TABLE 1.1.3. Summary of Waste Management Alternatives Described[a]

Waste Type	Fuel Cycle[b]	Waste Treatment	Interim Storage of Treated Wastes	Waste Transportation
Spent fuel	1	Packaging*	Unpackaged - water basin* Packaged - water basin[c] Packaged - dry caisson[c]* Packaged - surface cask[c] Packaged - air cooled vault[c]	Rail* Truck Barge
High-level liquid waste	2,3	Vitrification* Calcination	Liquid HLW[d] - tank* Solid HLW - water basin* Solid HLW - sealed cask[c]	Rail* Truck
Fuel residue	2,3	Package without compaction* Compaction of hulls Melting of hulls	Near-surface caisson[c]* Vault[c]	Rail* Truck
Failed equipment and noncombustible waste	2,3	Package with minimum treatment*	ILW[d] - outdoor and indoor*[c] LLW[d] - outdoor and indoor[c]	Rail* Truck
Combustible and compactable waste	2,3	Incineration* Minimum treatment	as ILW and LLW	
Degraded solvent	2,3	Incineration*	--	
Liquid wastes and particulate solids	2,3	Cementation* Bitumenization	as ILW and LLW	
Plutonium	2b	Conversion* to PuO_2	Vault	Rail* Truck
Off-gas particles	2,3	HEPA filtration*	as compactable waste	
FRP disolver off-gas	2,3	I removal (silver-loaded adsorbent) C removal (molecular sieve) Kr removal (cryogenic)* Combined I + C + Kr removal*	as LLW as LLW Gas cylinder in vault*	
Process off-gas	2,3	I removal (silver-loaded adsorbent)	as LLW	
Ventilation air	2,3	Group III filter/HEPA filter* Sand filter/HEPA filter Deep-bed fiber filter/HEPA filter	as compactable waste	

a. Asterisks denote components of the reference waste management system.
b. Fuel cycle 1 is the once-through case, fuel cycle 2 is the uranium-only recycle case (2b denotes the case in which the plutonium is purified and stored separately), and fuel cycle 3 is the plutonium plus uranium recycle case.
c. This system might be necessary only in the case of deferred repository availability.
d. HLW denotes high-level waste, ILW denotes intermediate-level waste, and LLW denotes low-level waste. A packaged waste surface dose rate of 0.2 R/hr is the dividing line between LLW and ILW in this report.

1.1.5 Secondary Wastes

Secondary wastes are wastes produced during management of the primary wastes. Such wastes generated in the reference integrated waste management system for the plutonium plus uranium recycle case are summarized in Table 1.1.4. Comparison of these data with those for the primary wastes (Table 1.1.1) shows the secondary waste volumes to be generally about 15% or less of that of the corresponding type of primary waste except for the miscellaneous liquid and particulate solid wastes; the volume of secondary TRU waste of this type is actually greater than the volume of primary TRU waste of this type. The greater volume is caused by the incineration of the primary combustible waste, which results in the generation of secondary waste off-gas scrubber solution as well as incinerator ash. An overall reduction in total waste volume does result from incineration, however, because the volume of incinerator ash and scrubber solution is much less than the volume of combustible waste.

TABLE 1.1.4. Secondary TRU Wastes from Reference Recycle System

Facility and Waste Type	Volume, m^3/MTHM	Radionuclide Content, Ci/MTHM[a]		
		Fission Products	Actinides	Activation Products
FRP				
Combustible and compactable wastes	0.28	3×10^{-1}	5×10^{-3}	9×10^{-3}
Miscellaneous liquid and particulate solid wastes	0.287	7	1×10^{2}	9×10^{-1}
Noncombustible wastes	0.075	1×10^{1}	2×10^{-1}	9×10^{-1}
MOX FFP				
Combustible and compactable wastes	0.006[b]	--	2×10^{-4}	--
Miscellaneous liquid and particulate solid wastes	0.069[b]	--	4×10^{1}	--
Noncombustible wastes	0.001[b]	--	1×10^{-3}	--

a. Ci/MTHM reprocessed; in the MOX FFP cases, multiply by 5 to obtain Ci/MTHM fabricated. Based on 1.5 yr-aged fuel at FRP and 1 yr after reprocessing for MOX FFP.
b. m^3/MTHM reprocessed. Multiply by 5 to obtain m^3/MTHM of MOX fuel fabricated.

1.1.6 Scope of Technology and Facility Descriptions

The bases for the facility design concepts and the way these concepts were developed are detailed in Section 3.6. Experts in each of the areas of waste management technology selected the design alternatives and the reference processes (from those identified as commercialized or available technology) and developed preliminary facility descriptions. An architect engineering firm completed the facility descriptions, developed the capital cost estimates, and estimated the construction requirements and impacts. This information along with results of evaluations of potential emissions and accidents, costs and safeguard requirements was then consolidated and presented in a standardized format, that included:

1. Process Alternatives
2. Facility Design Bases
3. Process Description
4. Facility Description
5. Operating and Maintenance Requirements
6. Secondary Wastes
7. Emissions
8. Decommissioning Considerations
9. Postulated Accidents
10. Costs
11. Construction Requirements
12. Effects of Fuel Cycle Options

1.1.7 Accident Analysis Basis

The primary emphasis in the analysis of accidents was to identify accidents with potential for offsite radionuclide releases; however, accidents with a potential for internal contamination or increased exposure to workers were also identified. An accident spectrum was developed which is believed to be generically representative of potential accidents in commercial radioactive facilities, although many of the postulated accidents might be eliminated by advanced plant design or operational techniques. Accident frequencies and source terms were estimated and the accident scenarios were classified into three accident severity groups: minor, moderate, or severe depending on the quantity of hazardous material released or the extent of increase of radiation fields in occupied zones.

1.1.8 Cost Analysis Basis

The radioactive waste management costs are detailed as capital cost, operating cost, and levelized unit costs for the various waste management facilities and systems:

- Capital costs are derived by estimating requirements for major equipment, buildings and structures, site improvements, and direct construction labor, and factoring these direct cost estimates to generate other direct costs as well as indirect construction costs, architect-engineer (A-E) costs, and owner's costs. Owner's costs consist of all costs incurred by the owner within his own organization in connection with the facility construction which normally do not form a part of the A-E or contractor scope of work. Owner's costs include such things as interest during construction, land acquisition costs, equipment spares, startup costs, insurance, etc. The sum of the above components is the reported total capital cost.
- Operating costs include labor, process materials, utilities, maintenance, overhead, and other miscellaneous items identified with the labor force or production. The number of man-hours, quantities of materials, and requirements for utilities have been derived in each case from the facility descriptions. The allowances for maintenance materials, overhead and miscellaneous costs have been derived by factoring either capital or direct labor costs.
- Levelized unit costs are calculated charges per unit of production sufficient to recover all capital including interest charges on debt and equity and pay all operating expenses including taxes and insurance. For this study, weighted costs of capital (interest charges on debt and equity) are estimated at 7 and 10% for government and private facilities, respectively. Unit cost ranges for government facilities corresponding to a range of cost of capital from 0 to 10% are also shown.

A constant dollar method of analysis is employed in which all costs, both present and future, are expressed in terms of the buying power of the dollar in mid-1976. Costs are escalated to a mid-1978 basis for this summary using a factor of 1.17. Both the 1976 and 1978 costs are shown in the summary.

1.1.9 Safeguards and Physical Protection Requirements Bases

Key features of safeguards and physical protection measures employed to prevent the willful release of radioactive material and to prevent the sabotage of nuclear facilities include definition of the threat, consideration of attractiveness and accessibility of the wastes and a description of the physical protection and safeguards measures that can be emplemented.

1.1.10 Fuel Cycle Projections

Both a reference nuclear power growth projection of 400 GWe installed in the year 2000 and a low-growth projection of 255 GWe installed in the year 2000 are developed. Schedules for installation of fuel cycle facilities that would be required for the implementation of both projections are presented.

1.2 WASTE TREATMENT TECHNOLOGY

1.2 WASTE TREATMENT TECHNOLOGY

Section 4 describes conceptual processes and facilities for treating gaseous and various transuranium (TRU) wastes produced during the post-fission portion of the LWR fuel cycle. The goal of the treatment process for TRU wastes and for long-lived radionuclides removed from the gaseous waste streams is to convert these wastes to stable products suitable for isolation in geologic repositories. Treatment concepts are based on available technology; they do not necessarily represent an optimum design but are representative of what could be achieved with current technology. In actual applications, these concepts probably will be improved, which might be reflected in either more effecient processes or lower environmental impacts or both. These conceptual descriptions provide a reasonable basis for cost analysis and for development of estimates of environmental impacts.

Technology areas for which waste treatment alternatives are described are:

Report Section	Section 4 Topics	Summary Section
4.1	High-level liquid waste solidification	1.2.1
4.2	Packaging of fuel residue	1.2.2
4.3	Failed equipment and noncombustible waste treatment	1.2.3
4.4	Combustible and compactable waste treatment	1.2.4
4.5	Degraded solvent treatment	1.2.5
4.6	Dilute aqueous waste pretreatment	1.2.6
4.7	Immobilization of wet and solid wastes	1.2.7
4.8	Off-gas particle removal systems	1.2.8
4.9	Fuel reprocessing plant dissolver off-gas treatment	1.2.9
4.10	Process off-gas treatment	1.2.10
4.11	Fuel reprocessing plant atmospheric protection system	1.2.11

In most cases more than one alternative is evaluated for each technology area. One alternative is selected as the reference system process to permit evaluation of secondary waste effects and other system impacts. In two of the areas, dilute aqueous waste pretreatment and off-gas particle removal systems, the descriptions are limited to generic discussions of facilities.

Application of waste treatment alternatives to TRU wastes is limited to the facilities (the fuel reprocessing plant and the mixed-oxide fuel fabrication plant) required to implement one of the fuel cycles involving recycle. This is because the wastes generated in the once-through fuel cycle (except for the spent fuel itself) are considered to be non-TRU wastes and thus outside the scope of this report. The management of spent fuel as a waste in the once-through fuel cycle primarily involves storage, transportation, and disposal of the spent fuel; the only operation that is considered to be "waste treatment" is packaging of the spent fuel. Because the conceptual packaging facility is designed as a modular addition to an independent spent fuel storage basin, this operation is described in the storage technology portion of the report (Section 5).

The plutonium plus uranium recycle system generates the most TRU waste because MOX FFPs are operated as well as FRPs. The amount of TRU waste generated at an FRP may also be greater in the

plutonium plus uranium recycle system than in the uranium-only recycle system, but this difference is not expected to be substantial. The treatment processes described in Section 4 are based on handling the wastes generated in the plutonium plus uranium recycle fuel cycle. Table 1.2.1 summarizes the quantities of packaged wastes resulting from the various reference system waste treatment processes; the information is correlated by waste container type and by the radiation dose rate at the surface of the containers. These waste quantities include the effects of secondary wastes generated during the treatment of the primary wastes. Table 1.2.2 summarizes the cost estimates developed for the various reference system waste treatment processes and Table 1.2.3 contains radionuclide release estimates developed for the operation of these processes.

Following these tables are brief descriptions of the various types of waste and of the treatment processes evaluated. Comparisons of major features of the reference system processes with those of other available alternatives are also given.

TABLE 1.2.1. Reference System Packaged Waste Output from 2000 MTHM/yr FRP and 400 MTHM/yr MOX FFP(a)

Waste Type	Location	Canisters/yr		Boxes/yr(b)		Drums/yr(c)		Volume, m^3/yr
		HLW(d)	ILW(e,f)	ILW(f)	LLW(g)	ILW(f)	LLW(g)	
High-level waste	FRP	657	--	--	--	--	--	145
Fuel residue	FRP	--	480	--	--	--	--	667
Failed equipment and noncombustible waste	FRP	--	71	--	60	4417	483	1363
	MOX FFP	--	--	--	20	--	394	164
Combustible and compactable waste	FRP	--	--	--	--	400	--	83
	MOX FFP	--	--	--	--	--	50	10
Wet waste and particulate solids	FRP	--	--	--	--	3355	2230	1172
	MOX FFP	--	--	--	--	--	1655	344
Dissolver off-gas	FRP	--	--	--	--	2(h)	46(h)	9
Vessel off-gas	FRP	--	--	--	--	(h)	(h)	--
Ventilation air	FRP	--	--	--	--	(h)	(h)	--

a. Based on recycle of plutonium and uranium, nonTRU wastes not included except in gaseous waste case.
b. 1.2 x 1.8 x 1.8 m (4.08 m^3)
c. 55-gal (0.208 m^3)
d. 30 cm in diameter and 3 m in length (0.22 m^3)
e. 75 cm in diameter and 3 m in length (1.39 m^3)
f. ILW containers are those non-HLW containers having surface dose rates in excess of 0.2 R/hr
g. LLW containers are those having surface dose rates below 0.2 R/hr
h. Filters generated here are included under compactable waste

TABLE 1.2.2. Costs of Reference System Waste Treatment Facilities and Operations

Waste Type	Location	1976 Dollars		1978 Dollars	
		Capital Cost, $1000	Levelized Unit Cost, $/kg HM[a]	Capital Cost, $1000	Levelized Unit Cost, $/kg HM[a]
High-level waste	FRP	47,000	8.90[b]	55,000	10.40
Fuel residue	FRP	15,000	4.20	17,000	4.90
Failed equipment and noncombustible waste	FRP	23,000	3.60	27,000	4.20
	MOX FFP	3,200	0.50	3,700	.60
Combustible and compactable waste	FRP	14,000	2.90	17,000	3.40
	MOX FFP	5,500	0.85	6,400	1.00
Wet waste and particulate solids	FRP	14,000	2.00	16,000	2.35
	MOX FFP	11,500	1.60	13,000	1.85
Dissolver off-gas	FRP	34,000	5.20	40,000	6.10
Vessel off-gas	FRP	23,000	3.30	27,000	3.85
Ventilation air	FRP	10,000	1.50	12,000	1.75

a. $/kg HM reprocessed. In the MOX FFP cases, multiply by five to obtain $/kg HM of MOX fuel fabricated. The uncertainties in the unit cost estimates are 40 ± 15%.
b. Does not include cost of lag storage and loadout facilities which are essential components of reference-system high level waste solidification (see Section 1.3.2.1).

TABLE 1.2.3. Radionuclide Releases Resulting from Reference System Waste Treatment Operations(a)

Waste Type	Location	Radionuclide Release, Ci/yr		
		Nonvolatile Fission Products[b]	Actinides	Nonvolatile Activation Products[c]
High-level waste	FRP	8×10^{-2}	8×10^{-8}	---
Fuel residue	FRP	8×10^{-7}	1×10^{-7}	$4\ 10^{-7}$
Failed equipment and noncombustible waste	FRP	6×10^{-6}	7×10^{-5}	2×10^{-12}
	MOX FFP	---	5×10^{-6}	---
Combustible compactable waste	FRP	2×10^{-7}	1×10^{-5}	8×10^{-11}
	MOX FFP	---	7×10^{-10}	---
Wet waste particulate solids	FRP	2×10^{-6}	2×10^{-6}	8×10^{-9}
	MOX FFP	---	9×10^{-12}	---
Dissolver off-gas	FRP	8	4×10^{-8}	---
Vessel off-gas	FRP	4	4×10^{-6}	---
Ventilation air	FRP	3×10^{-6}	4×10^{-7}	---

a. Released from the stack of the parent facility.
b. Nonvolatile fission products exclude ^{3}H, ^{85}Kr, and ^{129}I. During FRP operation with reference off-gas systems releases of these materials amount to about 7×10^{5} Ci/yr of ^{3}H, 2×10^{6} Ci/yr of ^{85}kr, and 7×10^{-2} Ci/yr of ^{129}I.
c. ^{14}C as CO_2 is a volatile activation product. During FRP operation with reference off-gas systems release of ^{14}C amounts to about 11 Ci/yr.

1.2.1 High-Level Waste Treatment

The high-level waste (HLW) stream of a typical nuclear fuel reprocessing plant is comprised primarily of the aqueous effluent from the first solvent extraction contactor. This effluent contains essentially all of the fission products and transuranic elements (except for plutonium) that are present in the spent fuel.

The assumption for the reference system is that the HLW is solidified within a short time after it is generated. An alternative approach involves tank storage for several years before solidification.

The reference system HLW solidification process is vitrification (conversion to glass). Another well developed alternative is calcination (conversion to oxides); this alternative is also described. Less well developed alternative solid waste forms are supercalcine, sintered glass, metal matrices, glass-ceramics, coated pellets, and ion exchange media.

The reference vitrification process is spray calcination/in-can melting. Alternative vitrification processes include the continuous melting process, the rotary kiln calcination/continuous melting process, the batch evaporation and melting process, and the direct-liquid-fed/continuous melting process.

The product of the reference vitrification process is a borosilicate glass containing about one part HLW oxides and two parts glass-forming additives by weight. It is a stable product resistant to water leaching. The vitrified product is contained in seal-welded stainless steel canisters 30 cm in dia and 3 m in length. Each canister contains the solidified high-level waste (SHLW) resulting from the processing of 3 MTHM of fuel. Repository limits on canister heat-generation rates may require a reduction in the quantity of SHLW in each canister either by dilution or by reducing the size of the canisters. Either method increases the number of canisters.

Continuous fluidized-bed calcination is the reference calcination process. Other calcination processes involve the use of pot calciners, spray calciners, or rotary kilns. The product of the reference calcination process is a thermally-stabilized, granular, free-flowing powder. The powder is sealed in stainless steel canisters 20 cm in dia and 3 m in length. A smaller diameter canister is used for waste calcine than for waste glass to prevent overheating at the center line of the canister (the thermal conductivity of calcine is approximately 25% that of glass). However, each process produces approximately the same number of canisters.

Table 1.2.4 provides some comparative data for the reference vitrification and calcination alternatives for a reprocessing rate of 2000 MTHM/yr.

Fifteen postulated accidents were examined for these high-level waste treatment alternatives. The most severe postulated atmospheric release amounts to that contained in 10^{-5} kg of calcine.

No special safeguards and physical protection requirements were identified for the solidification process. Key factors here are the protected location of the operations, the extremely high radiation level of the processed material, and the relatively low plutonium content of the material.

TABLE 1.2.4. Comparative Cost and Radionuclide Release Data for the Vitrification and Calcination Alternatives, Reprocessing Rate 2000 MTHM/yr

Alternative	Capital Cost, $1000(a)	Levelized Unit Cost, $/kg HM(a)	Package/yr	Packaged Waste Volume, m^3/yr	Total Nonvolatile Radionuclide Release, Ci/yr(b)
Vitrification	47,000 (55,000)	8.90 (10.40)	657	145	8×10^{-2}
Calcination	65,000 (76,000)	11.10 (13.00)	683	67	4×10^{-3}

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 30-35%.
b. Released from the stack of the parent facility.

1.2.2 Fuel Residue Treatment

Fuel residue waste material consists of the residue (fuel element hardware and chopped cladding material) remaining after the bulk of the fuel core material, including most of the actinides and fission products, has been dissolved in nitric acid in a chop and leach process. The residue is contaminated with low levels of actinides and fission products and contains essentially all the activation products formed in the hardware and cladding material.

In the reference system treatment process, the uncompacted fuel residues are dried then packaged under dry sand in seal welded stainless steel canisters. The sand insures that any phyrophoric Zircaloy fines will not catch fire. A material other than sand (e.g., helium) could be used as the package filler material. Each canister is 75 cm in dia by 3 m long and contains the fuel residue from the processing of about 4 MTHM of fuel.

Alternative processes include volume reduction of the cladding portion of the fuel residue by mechanical compaction or by melting. Press compaction was selected as the reference concept for mechanical compaction; other concepts considered were high-energy rate compaction, extrusion, swaging, and flattening. The Inductoslag process was selected as the reference melting concept; in this process the melt is inductively heated and is insulated from a water-cooled crucible by a layer of frozen slag. Other melting concepts considered required different crucible materials and/or heating methods.

The press compaction process produces 97-kg compacts 23 cm in dia by 0.71 m long. The melting process produces 390-kg ingots 23 cm in dia by 1.45 m long. Twenty-eight compacts or fourteen ingots can be packaged in the same size canister as that used for the uncompacted residue. The packaged waste volumes resulting from the three alternative processes described in this report are compared in Table 1.2.5 along with the costs and radionuclide releases.

Two postulated accidents were examined for the fuel residue treatment processes. The quantity of tritium contained in 1 to 10 kg of hulls is postulated to be released in these accidents but no significant release of nonvolatile radionuclides is expected.

No special safeguards and physical protection requirements were identified for the fuel residue treatment processes. The plutonium content is quite low, the radiation level is high enough that special handling is required, and the operations are conducted in protected locations.

TABLE 1.2.5. Comparative Cost and Radionuclide Release Data for Fuel Residue Treatment Alternatives, Reprocessing Rate 2000 MTHM/yr

Alternative	Capital Cost, $1000(a)	Levelized Unit Cost, $/kg HM(a)	Packaged Waste Volume, m^3/yr	Total Nonvolatile Radionuclide Release, Ci/yr(b)
Packaging without compaction	15,000 (17,000)	4.20 (4.90)	670	1×10^{-6}
Compaction of hulls	17,000 (20,000)	3.90 (4.60)	400	1×10^{-6}
Melting of hulls	23,000 (27,000)	4.40 (5.20)	270	1×10^{-6}

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 25-35%.
b. Released from the stack of the parent facility.

1.2.3 Treatment of Failed Equipment and Noncombustible Waste

The generation of small items of noncombustible waste is more or less routine and predictable at an FRP or a MOX FFP. The failure of large items of process equipment, while not routine, will certainly occur. Failed equipment will require rapid removal and replacement to minimize its effect on plant efficiency. Metal is the primary constituent of failed equipment and noncombustible waste, but substances such as glass and concrete are also present.

Reference treatment of these wastes involves packaging them either in 55-gal drums, 1.2 x 1.8 x 1.8 m steel boxes, or canisters like those used for the packaging of fuel residue. Treatment may also involve decontamination and/or disassembly prior to packaging. The reference system includes decontamination of failed equipment (but not of noncombustible waste) and disassembly of the failed equipment items that cannot be packaged directly. Only the reference concept is described because there are no significant alternatives available. Data describing the output, the costs, and the emissions for this treatment process are contained in Tables 1.2.1 through 1.2.3.

Two postulated accidents were examined for the failed equipment and noncombustible waste treatment process. Neither accident results in any radionuclide release.

No special safeguards and physical protection requirements were identified for this process. The plutonium content is quite low for failed equipment and noncombustible waste, and the operations are conducted in protected locations.

1.2.4 Treatment of Combustible and Compactable Waste

These waste materials include ventilation filters, which remove suspended particles from gaseous streams; severely degraded extractant, which could result from operational malfunctions; and ion exchange resins, which may be generated during product or extractant purification steps. General combustible trash, which consists primarily of cloth, paper, wood, plastics and rubber is the major part of these waste materials.

The reference system treatment process involves incineration of the combustibles and compaction of the ventilation filter glass-fiber media. Two incinerators are used in the FRP, one to process LLW and one to process ILW; one incinerator is used in the MOX FFP. These are dual chamber incinerators. They have a limited air supply in the lower chamber to maintain quiescent burning. An excess of air and additional propane in the upper chamber thoroughly burn the gases evolved from the lower chamber. The off-gases are scrubbed for removal of acidic gases and particulates, and the scrubbing solution (after prior concentration in the case of the FRP) is sent along with the incinerator ash to the wet waste and particulate solids immobilization facility.

Ventilation filters are treated in the reference system by punching out and pelletizing the filter media, packaging the pelletized media in drums (along with metal filter frames, if they are present), and shredding and burning wooden filter frames.

The alternative treatment process that was evaluated involves packaging the combustible and compactable waste in drums except for the severely degraded extractant, which is burned in a special incinerator. This alternative is referred to as minimum treatment. Treatment by compaction and by shredding followed by immobilization in a solid matrix were also considered. Table 1.2.6 compares features of the incineration and minimum treatment alternatives.

Eight postulated accidents were examined for these combustible and compactable waste treatment alternatives. The most severe postulated release of nonvolatile radionuclides amounts to about 10^{-7} of the annual input. This release could occur if a fire that is severe enough to result in failure of the cell HEPA filters started in the feed preparation line.

No special safeguards or physical protection requirements were identified for these waste treatment processes unless the drums containing ventilation filters from the MOX FFP are distinguishable from the other waste drums. The drums containing the MOX FFP filters might contain up to 200 g of Pu per drum but the number of such drums would be only about 2% of the total number; this results in a target considered to be too diffuse to be a likely candidate for theft as these drums are not distinguishable from others.

1.2.5 Degraded Solvent Treatment

Solvent that is degraded to the extent that purification by the normal treatment system is impossible or impractical is a waste requiring disposal. A separate solvent incinerator is described for this purpose. Such an incinerator would be needed should a treatment process other than incineration be chosen for treatment of the other combustible wastes. The reference solvent incinerator operation atomizes the feed into a forced-convection incineration unit. Another approach that was considered involves burning the solvent on a quiescent liquid surface with a second-stage burner to complete the oxidation of organic vapors and unburned carbon.

TABLE 1.2.6. Comparative Cost and Radionuclide Release Data for Combustible and Compactible Waste Treatment Alternatives, Reprocessing Rate 2000 MTHM/yr

Alternative	Capital Cost, $1000(a)	Levelized Unit Cost, $/kg HM(a,b)	Packaged Waste Volume, m^3/yr	Total Nonvolatile Radionuclide Release, Ci/yr(c)
ILW at FRP				
Incineration	8,800 (10,000)	1.90 (2.20)	345(d)	1×10^{-5}
Minimum treatment plus Solvent incineration	16,000 (19,000)	3.10 (3.60)	2375	1×10^{-8}
LLW at FRP				
Incineration	5,500 (6,400)	1.00 (1.20)	380(d)	6×10^{-12}
Minimum treatment	6,000 (7,000)	0.90 (1.10)	2500	1×10^{-13}
LLW at MOX FFP				
Incineration	5,500 (6,400)	0.85 (1.00)	235(d)	7×10^{-10}
Minimum treatment	2,500 (2,900)	0.35 (0.40)	328	2×10^{-12}

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 35%.
b. $/kg HM reprocessed. In the MOX FFP cases, multiply by five to obtain $/kg HM of MOX fuel fabricated.
c. Released from the stack of the parent facility.
d. Pelletized filter media plus incinerator ash and off-gas scrubbing solution immobilized in cement. Neither costs incurred nor emissions from cementation included here.

The capital cost of the reference forced-convection unit is estimated at $8,000,000 and the levelized unit cost is estimated at $1.40/kg HM (mid-1978 dollars). The atmospheric release of nonvolatile radionuclides is estimated to be 4×10^{-9} Ci/yr. These amounts were included in the values listed in Table 1.2.6 for the minimum treatment plus solvent incineration case.

Five postulated accidents were examined for the solvent incinerator. The most severe postulated release of nonvolatile radionuclides amounts to about 10^{-8} of the annual feed activity.

No special safeguards are required for the solvent incinerator because of the solvent's low radionuclide content. In addition, the process would be carried out within the FRP, which is well safeguarded.

1.2.6 Dilute Aqueous Waste Pretreatment

The initial step for treating most aqueous waste streams is some type of concentration process that converts the liquid waste feed material into two product streams: 1) a concentrated liquid, slurry, or particulate solid containing the bulk of the radionuclides and nonradioactive

chemicals and 2) a purified liquid containing significantly reduced contaminant concentrations. The processes utilized for these initial concentration steps include evaporation, ion exchange, filtration or centrifugation, reverse osmosis, and electrodialysis. Only brief generic descriptions of these processes are provided because they are more closely related to main plant operation than to waste treatment. The concentrated waste streams produced by these processes require additional waste treatment, which is discussed in the following section.

1.2.7 Treatment of Wet Wastes and Particulate Solids

In the reference system the wet and particulate solid TRU wastes include: 1) the concentrated aqueous intermediate-level liquid waste (ILLW) stream resulting from the final solvent extraction purification cycles, 2) solid materials used in the final purification and the fluorination of the uranium product, and 3) waste materials resulting from incineration of combustible wastes. The ILLW could be deleted from this list by solidifying it with the high-level liquid waste. Ion exchange resins, which in the reference system are incinerated, could be added to the list as an alternative to their incineration.

These wastes are to be immobilized in a liquid free form before their ultimate disposition. The reference system immobilization process is cementation in steel drums using a drum-tumbling system. This process involves mixing wastes with cement in drums and allowing the mixture to harden to a liquid-free product. In-drum mixers and in-line mixers are also available. The alternative approach selected for evaluation is bitumenization by a continuous screw extruder process. Batch stirred-evaporator and continuous turbulent film evaporator bitumenization processes are also available. Other alternative approaches not evaluated include immobilization by a urea-formeldahyde process, packaging with various absorbents to immobilize the water, and immobilization by various polymeric solidification processes.

The output of a wet waste and particulate solids immobilization facility depends to a considerable extent on the process chosen for treatment of the combustible waste. Table 1.2.7 shows comparative data for cementation and bitumenization with incineration or minimum treatment of combustible waste. These data are for a reprocessing rate of 2000 MTHM/yr.

The packaged waste volume is markedly lower in the bitumenization case than in the cementation case. This is because the water contained in the waste is driven off by the bitumenization process but not by the cementation process.

Eight postulated accidents were examined for these immobilization processes. The most severe postulated atmospheric release of nonvolatile radionuclides amounts only to that contained in about 10^{-6} kg of fixed waste; this postulated accident involves cell HEPA filter failure during a bitumen fire.

No special safeguards or physical protection requirements were identified for these immobilization processes. The treated wastes contain relatively little plutonium or other radionuclides and the treatment processes make them more difficult to recover. In addition, the processes would be conducted within the FRP, which is well safeguarded.

TABLE 1.2.7. Comparative Cost and Radionuclide Release Data for Cementation and Bitumenization with Incineration or Minimum Treatment of Combustible Waste, Reprocessing Rate 2000 MTHM/yr

Alternative	Capital Cost, $1000[a]	Levelized Unit Cost, $/kg HM[a,b]	Packaged Waste Volume, m^3/yr	Total Nonvolatile Radionuclide Release, Ci/yr[c]
Combustible Waste Incineration				
Cementation				
at FRP	14,000 (16,000)	2.00 (2.35)	1172	4×10^{-6}
at MOX FFP	12,000 (14,000)	1.60 (1.85)	344	9×10^{-12}
Bitumenization				
at FRP	14,000 (16,000)	2.00 (2.35)	534	4×10^{-6}
at MOX FFP	12,000 (14,000)	1.60 (1.85)	112	9×10^{-12}
Combustible Waste Minimum Treatment				
Cementation				
at FRP	14,000[d] (16,000)	2.00[d] (2.35)	530	3×10^{-6}
at MOX FFP	12,000[d] (14,000)	1.60[d] (1.85)	123	2×10^{-12}
Bitumenization				
at FRP	14,000[d] (16,000)	2.00[d] (2.35)	291	3×10^{-6}
at MOX FFP	12,000[d] (14,000)	1.60[d] (1.85)	97	2×10^{-12}

a. Mid-1976 dollars (mid-1978 dollars in parenthesis). The uncertainty in the unit cost estimates is 35%.
b. $/kg HM reprocessed. In the MOX FFP cases, multiply by five to obtain $/kg HM of MOX fuel fabricated.
c. Released from the stack of the parent facility.
d. Assuming same immobilization facility as in the incinerated mode.

1.2.8 Off-Gas Particle Removal Systems

Off-gas streams from processes and operations associated with fuel storage, fuel reprocessing, and mixed-oxide fuel fabrication contain radionuclide-contaminated particles. These particles require removal before the off-gas streams can be released to the atmosphere.

Prefilters are used to remove most airborne particles larger than 10 μm. Available prefilter shapes, sizes, filter media, and removal efficiencies are discussed. High-efficiency particulate air (HEPA) filters, which are used for most efficient removal of smaller particles, are also discussed. Removal efficiencies of 90% by a prefilter and 99.9% by each stage of HEPA filtration were used to estimate particulate radionuclide releases in this report. These removal efficiencies are lower than those usually measured; therefore, their use in estimating releases provides a degree of conservatism to the release estimates.

Alternative particle removal systems considered were electrostatic precipitators, bag filters, and wet collectors. These systems are used in normal industrial air cleaning applications but have not found extensive application in nuclear facilities.

Special safeguards and physical protection measures may be required for filters removed from glove boxes where PuO_2 is processed.

1.2.9 FRP Dissolver Off-Gas Treatment

Operation of a fuel reprocessing plant can result in release to a gaseous stream of most if not all the fission product tritium, krypton, and iodine and also the activation product carbon-14. Important quantities of fission product ruthenium can also enter a gaseous stream.

The reference FRP operation results in the release to the atmosphere of virtually all the tritium present in the spent fuel. Essentially all the tritium is present as tritiated water and is released from the plant in a vaporized excess water stream.

The dissolver off-gas stream from the dissolution of spent fuel contains practically all the krypton, iodine, and carbon present in the spent fuel. The stream also contains a small fraction of the ruthenium. The reference dissolver off-gas treatment system removes the bulk of these materials from the off-gas stream. The ruthenium is first removed by adsorption on silica gel, and the iodine is then removed by adsorption on a silver-loaded adsorbent. After removal of residual oxides of nitrogen (in a catalytic reactor) and water vapor, CO_2 is removed from the off-gas by molecular sieves (Zeolite beds); the sieves are periodically regenerated and the carbon-14 is ultimately recovered as calcium carbonate. Finally, the krypton is removed from the off-gas by a cryogenic fractional distillation process, yielding a final product containing 80% krypton; this final product is placed in high-pressure gas cylinders for storage. The iodine-loaded adsorbent and the calcium carbonate are packaged in drums and treated like TRU wastes because of their long half-lives (1.6×10^7 yr for ^{129}I and 5700 yr for ^{14}C). The ruthenium loaded adsorbent after being packaged in steel drums is sent to a burial ground for non-TRU wastes.

Alternative methods of iodine-129 removal include scrubbing the gas stream with aqueous solutions such as caustic, mercury nitrate-nitric acid, or 20 to 22 molar HNO_3. Alternative methods of carbon-14 removal include caustic scrubbing and absorption in a liquid fluorocarbon. An alternative method of krypton removal is by absorption in a liquid fluorocarbon. None of these alternatives are described in this report because of either their low efficiency or their incomplete development.

Data on the combined reference dissolver off-gas treatment system are contained in Tables 1.2.1 - 1.2.3. Also considered were the recovery of only iodine, of carbon plus iodine, and of krypton plus iodine, (using the same processes as in the combined system). The cost data for these alternatives are compared in Table 1.2.8. The cost of the three separate alternatives is greater than that of the combined system because of some identical features of the carbon and krypton removal processes.

Fifteen postulated accidents were examined for the dissolver off-gas treatment processes. Most of the accidents that resulted in radionuclide releases involve short-term shutdown of a treatment process, thus allowing a short-term atmospheric release of a volatile radionuclide if

TABLE 1.2.8. Comparative Cost Data for Alternative Dissolver Off-Gas Treatment Systems, Reprocessing Rate 2000 MTHM/yr

Elements Removed	Capital Cost, $1000[a]	Levelized Unit Cost, $/kg HM[a]
I + C + Kr	34,000 (40,000)	5.20 (6.10)
I only	11,000 (13,000)	1.70 (2.00)
C (following I)	7,000 (8,000)	1.00 (1.20)
Kr (following I)	22,000 (26,000)	3.40 (4.00)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 40%.

the shutdown occurs during fuel dissolution. One accident, an explosion in an oxygen recombiner, is postulated to release 53 Ci of Kr-85 at ground level.

The only special physical protection requirement identified for the dissolver off-gas treatment processes involves the krypton stored in gas cylinders. An uncontrolled release of this concentrated material could be hazardous and precautions are necessary to prevent this from occurring.

1.2.10 Process Off-Gas Treatment

A separate system, the vessel off-gas system, is included in the FRP to serve process vessels and equipment other than the dissolver and shear. This system removes iodine, again by adsorption on a silver-loaded adsorbent, and filters process gas streams to remove particulate material. The estimated costs of and emissions from this facility are included in Tables 1.2.2 and 1.2.3.

Three postulated accidents were examined for the vessel off-gas treatment process. The only one resulting in radionuclide release involves a shutdown of the off-gas system.

A similar process off-gas system to treat the off-gas from cask venting and leaking fuel assemblies at an ISFSB is also described.

No special safeguards and physical protection requirements were identified for these process off-gas treatment systems.

1.2.11 FRP Atmospheric Protection System

Another gaseous waste treatment process is the atmospheric protection system (APS) used to provide final filtration of the ventilation air. Three systems that differ only in the type of prefilter used before the final passage through high-efficiency particulate air (HEPA) filters are described. The reference system uses Group III prefilters, which are extended-medium, dry, throw-away type prefilters. The other alternatives described are the sand filter and the deep-bed glass fiber filter systems. Table 1.2.9 compares the costs of these three alternative prefilter/HEPA filter systems; all of the systems provide a DF of 10^4 for particulates.

TABLE 1.2.9. Comparative Cost Data for Alternative Prefilter Systems, Reprocessing Rate 2000 MTHM/yr

Prefilter Alternative	Capital Costs, $1000[a]	Levelized Unit Cost, $/kg HM[a]
Group III Filter Module	10,000 (12,000)	1.50 (1.75)
Sand Filter	24,000 (28,000)	3.20 (3.75)
Deep Bed Fiber Filter	11,000 (13,000)	2.15 (2.50)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 35-40%.

Four postulated accidents were examined for the APS alternatives. The largest postulated radionuclide release amounts to 0.01 Ci; this release occurs from the rupture of a final HEPA filter.

No special safeguards and physical protection requirements were identified for the APS alternatives.

1.3 INTERIM STORAGE TECHNOLOGY

1.3 INTERIM STORAGE TECHNOLOGY

Section 5 describes conceptual facilities for interim storage of various treated transuranic (TRU) and gaseous waste products produced during fuel reprocessing and mixed-oxide fuel fabrication. Alternatives for interim storage of spent fuel prior to reprocessing or geologic isolation are also described. The storage concepts are based on available technology. They do not necessarily represent optimum designs, but are representative of what could be achieved with current capabilities. In actual applications some improvements could probably be made in these concepts, resulting in more efficient operation, lower environmental impacts, or both. These conceptual descriptions provide a reasonable basis for cost analysis and for development of estimates of environmental impacts.

In this summary, the interim storage technologies are discussed as they are combined in waste management system applications, rather than the order in which they are described in Section 5. This was done to facilitate the explanation of the function and importance of each technology. The contents of Section 5 and location where each component is discussed in this summary are as follows:

Report Section	Section 5 Topics	Summary Section
5.1	High-level liquid waste storage	1.3.2.1
5.2	Fuel residue storage	1.3.2.3
5.3	Other transuranic non-high-level solid waste storage	1.3.2.4
5.4	Solidified high-level waste storage	1.3.2.1
5.5	Plutonium dioxide storage	1.3.2.5
5.6	Krypton storage	1.3.2.2
5.7	Storage and packaging of spent fuel	1.3.1

Implementation of many of these alternatives will depend on the timing of geologic repository availability as well as on the fuel cycle chosen.

1.3.1 Interim Storage in the Once-Through Cycle

The reference once-through fuel cycle includes two definite waste management requirements, 1) storage of unpackaged fuel in water basins (either nuclear power plant basins or independent storage basins; only storage in independent basins is evaluated here), and 2) packaging of the spent fuel for the final repository. If a repository is not available at the time the fuel is ready for final isolation, the packaged fuel will require interim storage until the repository is available. Dry caisson storage is assumed. Cost and emissions data for these reference alternatives are summarized in Table 1.3.1. The following paragraphs contain more information on these alternatives and also provide comparisons with alternatives studied but not included in the reference system.

TABLE 1.3.1. Interim Storage in the Once-Through Cycle Reference System

Storage Operation	Facility Capacity, MTHM		Capital Cost, $1000s[a]	Levelized Unit Cost, $/kg HM[a]		Nonvolatile Radionuclide Release, Ci/yr[b]	
	Receiving	Storing		Privately Owned[c]	Federally Owned[c]	Fission Products	Activation Products
Reference System							
Unpackaged fuel in independent storage basins	2000/yr	3,000	220,000 (260,000)	113.3 (132.6)	57.3 (67.0)	5×10^{-2}	7×10^{-3}
Fuel packaging	2000/yr	--	109,000 (130,000)	25.3[d] (29.6)	15.6[d] (18.2)	1×10^{-2}	3×10^{-4}
Added to Reference System in Case of Deferred Repository Availability							
Packaged fuel in dry caisson	2000/yr	20,000	290,000 (340,000)	--	19.0 (22.2)	--	--

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainties in the unit cost estimates are 25-35%.
b. Released from full facility. Nonvolatile radionuclides exclude ^{3}H, ^{14}C, ^{85}Kr, and ^{129}I.
c. Costs for privately owned facilities are higher than Federally owned facilities because of taxes and a higher cost of money.
d. Includes incremental cost of modifying an independent storage basin to service a contiguous packaging facility.

1.3.1.1 Storage of Unpackaged Fuel

After the fuel is discharged from the nuclear power plant, it is stored to allow short-lived radionuclides to decay, thus decreasing the radiation level and the heat generation rate. Except for the first 0.5 year of storage which is assumed to always be in the power plant basin, this interim storage of the unpackaged fuel can be done at either the power plant basin or at an independent spent fuel storage facility (ISFSF), also sometimes referred to as "away from reactor" or AFR storage. The reference process for the short-term storage of unpackaged spent fuel is the water basin storage method, used successfully for over 20 years. Storage in air-cooled vaults was also considered but not in detail. For the reference system, short-term storage for 6.5 years is assumed before the fuel is packaged for the final repository or for longer term storage.

The reference ISFSF has a storage capacity of 3000 MTHM. Two types are described, 1) a separately located one having the capacity to ship and receive fuel at rates up to 500 MTHM/yr, and 2) one that is colocated with a fuel packaging facility and has the capacity to receive up to 2000 MTHM/yr. Cost and emission estimates for the latter type are included in Table 1.3.1. The capital cost of the first type of ISFSF is lower by $24 million, but the levelized unit costs are taken to be the same in the two cases, and the incremental costs are assigned to the fuel packaging operation.

Fourteen postulated accidents were examined for the water basin storage of unpackaged spent fuel. None result in the atmospheric release of any nonvolatile radionuclides. Krypton-85 is the predominant volatile radionuclide that will be released as a result of accidents; the largest postulated release is 5000 Ci, which would result from a criticality accident.

No special physical protection and safeguard requirements beyond those required for vital facilities and special nuclear materials access areas were identified for this technology.

1.3.1.2 Packaging of Spent Fuel

In the reference packaging process, cleaned and dried fuel assemblies are individually packaged in helium-filled steel (stainless steel if subsequent storage will be in water) containers that are then seal welded.

The spent fuel packaging facility is assumed to be colocated with an ISFSF that has been modified to increase its receiving capacity to 2000 MTHM/yr to match the packaging facility capacity. Cost and emission estimates for the packaging facility are contained in Table 1.3.1.

Six postulated accidents were examined for the spent fuel packaging facility. The largest atmospheric release of radionuclides is expected to result from dropping a fuel assembly; should 20% of the fuel rods rupture, a release of 2×10^{-8} Ci of nonvolatile radionuclides and 180 Ci of ^{85}Kr and other volatile radionuclides is estimated.

No special physical protection and safeguards requirements beyond those required for vital facilities and special nuclear materials access areas were identified for this technology.

1.3.1.3 Interim Storage of Packaged Fuel

Interim storage of packaged fuel will be necessary if final waste isolation facilities are not available at the time of packaging. The reference concept for storage of packaged fuel is the dry caisson concept. In this concept the packaged fuel is placed in a below grade steel caisson, which is closed with a concrete plug. This concept relies on the soil to conduct the radioactive decay heat from the spent fuel to the earth's surface, where it is dissipated to the atmosphere. Alternative packaged fuel storage concepts that were also evaluated are water basin storage, natural-draft air-cooled vault storage, and surface cask storage. All of these concepts assume facility expansion in 2000 MTHM modules up to an ultimate capacity of 20,000 MTHM. Costs of these alternative storage methods are compared in Table 1.3.2 (Federal ownership is assumed).

TABLE 1.3.2. Estimated Costs for Interim Storage of Spent Fuel

Storage Method	Capital Cost, $1000s[a]	Levelized Unit Cost, $/kg HM[a]
Dry caisson	290,000 (340,000)	19.0 (22.2)
Surface Cask	220,000 (260,000)	25.8 (30.2)
Air-cooled vault	510,000 (600,000)	29.8 (34.9)
Water basin	250,000 (300,000)	32.8 (38.4)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainties in the unit cost estimates are 20-25%.

Costs are based on the assumption that the packaged fuel storage facilities are located on the same site as the packaging facility. If the interim storage facility is located at a site separate from the packaging facility, then an independent site receiving and shipping facility must be colocated with the interim storage facility. This receiving facility has the capacity to receive or ship a total of 2000 MTHM/yr. The capital cost of such a facility is estimated to be $79 million and the levelized unit cost is estimated to be an additional $8.10/kg HM (assuming Federal ownership).

Eighteen postulated accidents were examined for these packaged fuel storage technologies. The largest atmospheric release is estimated to occur during a criticality incident in a water basin; the expected releases from such an incident would be no greater than those described earlier for packaged fuel storage.

No special physical protection and safeguard requirements beyond those required for vital facilities and special nuclear materials access areas were identified for these packaged fuel storage technologies.

1.3.2 Interim Storage for Reprocessing Cycle Wastes

The reference plutonium plus uranium recycle system includes two definite interim storage requirements 1) five-year storage of solidified high-level waste (SHLW) in a water basin, and 2) storage of the separated krypton. If a repository is not available at the time the wastes are ready for final isolation, interim storage will also be required for fuel residues and for the other transuranic non-high-level solid wastes; extended storage of SHLW will also be needed. Cost and emissions data for these reference alternatives are summarized in Table 1.3.3. The following paragraphs contain more detailed information on the reference alternatives and also provide comparisons with alternatives studied but not included in the reference system.

Interim storage requirements for a uranium-only recycle system would be essentially identical to that for the plutonium plus uranium recycle case except for the possible addition of plutonium dioxide storage facilities. Such a facility is also described.

1.3.2.1 High-Level Waste Storage

In the reference system high-level liquid waste (HLLW) is solidified immediately followed by storage for cooling purposes of the packaged solidified high-level waste (SHLW) for 5 years in a water basin adjacent to the FRP. The reference water basin facility thus has the capacity to hold the SHLW resulting from reprocessing 10,000 MTHM of fuel. The SHLW packages are then shipped to a geologic repository or to an interim storage site if a repository is not available.

The reference concept for use at an interim storage site is the sealed storage cask concept. In this concept, the SHLW package is placed in a high-integrity metal cask that is contained in a reinforced concrete radiation shield. Air circulating by natural convection between the shield and the sealed cask removes heat that is generated within the SHLW. The reference sealed storage cask facility has an initial capacity to store 2000 canisters of SHLW (the waste from reprocessing 6,100 MTHM) and can be expanded in 2000 canister modules to an ultimate capacity of 20,000 canisters.

TABLE 1.3.3. Interim Storage in the Plutonium Plus Uranium Recycle Reference System

Waste Type	Storage Method	Location[d]	Capital Cost, $1000s[a]			Levelized Unit Cost, $/kg HM[a,b]			Nonvolatile Radionuclide Release, Ci/yr[c]	
			Case A[e]	Case B[f]	Case C[g]	Case A	Case B	Case C	Fission Products	Actinides
Reference System										
Solidified HLW	Water basin	FRP	--	--	55,000	--	--	7.80	1×10^{-4}	8×10^{-7}
	Shipping facility	FRP	--	--	29,000	--	--	4.00	7×10^{-5}	6×10^{-8}
					84,000 (98,000)			11.80 (13.80)	2×10^{-4}	9×10^{-7}
Krypton	Gas cylinder	FRP	--	--	160,000 (190,000)	--	--	14.00 (16.40)	--	--
Added to Reference System in Case of Deferred Repository Availability										
Solidified HLW	Sealed cask	Independent	91,000 (106,000)	99,000 (115,000)	--	28.00[h] (32.80)	13.80[h] (16.20)	--	--	--
Fuel residue	Near surface	FRP	--	--	35,000 (41,000)	--	--	10.50 (12.30)	--	--
		Independent	87,000 (102,000)	167,000 (195,000)	--	5.40 (6.30)	5.30 (6.20)	--	--	--
Intermediate level solid waste	Shielded building	FRP	--	--	16,000 (19,000)	--	--	4.40 (5.15)	--	--
		Independent	47,000 (55,000)	83,000 (97,000)	--	2.30 (2.70)	2.20 (2.60)	--	--	--
Low level solid waste	Outdoor surface	FRP	--	--	1,100 (1,300)	--	--	0.35 (0.40)	--	--
		MOX FFP	--	--	1,000 (1,200)	--	--	0.32 (0.38)	--	--
		Independent	3,900 (4,600)	5,700 (6,600)	--	0.30 (0.35)	0.25 (0.30)	--	--	--

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainties in the unit cost estimates range from 20 to 40%.
b. $/kg HM reprocessed. In the MOX FFP case, multiply by five to obtain $/kg HM of MOX fuel fabricated.
c. Released from full facility. Nonvolatile radionuclides exclude H-3, C-14, Kr-85 and I-129.
d. FRP and MOX FFP locations are privately owned and operated. Independent locations are federally owned and operated.
e. Case A provides for storage of wastes generated by all FRPs and MOX FFPs operating in the reference system through 1990.
f. Case B provides for storage of wastes generated by all FRPs and MOX FFPs operating in the reference system through 1995.
g. Case C provides for onsite storage of the wastes generated during the first five years of reference facility operation.
h. These values are for a lower fuel throughput than the others because of the five year storage in the water basin prior to use of this method.

1.3.5

Data for these two stages of SHLW interim storage are included in Table 1.3.3. No other alternatives were evaluated in detail, but the alternatives discussed in Section 1.3.1.3 for packaged fuel could be used for SHLW as well.

Twenty postulated accidents were examined for these solidified high-level waste storage alternatives. The most severe atmospheric release is estimated to involve only 0.003 Ci of mixed fission products. This release is postulated to result from a tornado striking the water basin storage facility.

The physical protection and safeguard requirements of the FRP are adequate also for the water basin storage of solidified high-level waste. The massive enclosures of reinforced concrete used in the sealed cask method of storage provide substantial protection by themselves; additional protection would be provided by security personnel at the storage site.

An alternative to immediate solidification of HLLW followed by storage of the SHLW for cooling purposes is to store the HLLW for an extended period before the HLLW is solidified. If this alternative is chosen, the preferred method would almost certainly involve the storage of acidic HLLW in large subsurface stainless steel tanks. This concept was examined in detail. The cost data are summarized in Table 1.3.4.

TABLE 1.3.4. Estimated Interim Storage Costs for HLLW

	Capital Cost, $1000s[a]	Levelized Unit Cost, $/kg HM[a]
4 - year storage capacity	240,000 (280,000)	33.0 (39.0)
5 - year storage capacity	271,000 (317,000)	36.0 (42.0)

a. Mid-1976 dollars (mid-1978 dollars in parenthesis). The uncertainty in the unit cost estimate is 30%.

The storage of liquid HLW is clearly an expensive alternative compared to storage of solidified HLW (Table 1.3.3). In either case, the shipping facility is also required and its costs (Table 1.3.3) must be factored into the overall picture.

Nonvolatile radionclide releases to the atmosphere from the filled HLLW storage facilities are estimated to be 1×10^{-3} Ci/yr of fission products and 2×10^{-6} Ci/yr of actinides.

Eight postulated accidents were examined for the tank storage of HLLW. The most severe postulated atmospheric release amounts to less than 10^{-7} kg of HLW calcine equivalent.

The tank design features employed to provide safe containment of this intensely radioactive solution are such that additional physical protection measures are not required to guard against theft or sabotage.

1.3.2.2 Krypton Storage

The high-pressure gas cylinders containing the krypton recovered from the dissolver off-gas stream of the FRP are stored in a special facility constructed adjacent to the FRP. In the reference facility, gas cylinders are stored in cells that are cooled with refrigerated air,

which removes the radioactive decay heat. Cylinder storage is planned for at least 50 years, after which the krypton (then containing less than 4% of the initial amount of ^{85}Kr) could be released to the atmosphere.

The reference facility is designed to contain 10 years' product from the reference FRP (about 145 cylinders/yr), with provisions for expansion every 10 years until a total of 30 years' storage capacity is reached at which time the FRP is assumed to be shut down. The costs associated with krypton storage are included in Table 1.3.3. No other krypton storage concept was examined in detail.

Four postulated accidents were examined for this krypton storage technology. The most serious involves rupture of a gas cylinder in the operating area or storage corridor. Should this occur, up to 10^5 Ci is released to the stack and workers in the vicinity of the rupture may receive significant exposures.

The physical protection features of the storage building itself and of the FRP site at which it is located are expected to provide adequate protection against sabotage.

1.3.2.3 Fuel Residue Storage

If a repository is not available, the reference storage method for these wastes is storage of the fuel residue canisters in near-surface caissons. The caissons are positioned vertically in an engineered, above-grade soil structure and are capped with a steel and concrete plug.

An alternative storage concept was also evaluated in which the waste canisters were stored inside sleeves in a concrete vault. Cost data were developed for the two alternative methods for storing canisters of fuel residue (or for storing similar canisters containing failed equipment) for 1) the case in which the canisters of fuel residue and failed equipment that were filled during the first five years of one FRP operation are stored at the FRP site, 2) the case in which the canisters that were filled during operation of all FRPs through the year 1990 are stored at an independent federal site and 3) storage at an independent federal site of the canisters that were filled during operation of all FRPs through the year 1995. These data are compared in Table 1.3.5.

TABLE 1.3.5. Estimated Interim Storage Costs for Fuel Residues and Similarly Packaged Failed Equipment

Storage Requirement	Location	Capital Cost, $1000s[a]		Levelized Unit Cost, $/kg HM[a]	
		Near-Surface	Vault	Near-Surface	Vault
First 5 years' output of 1 FRP	FRP	35,000 (41,000)	120,000 (140,000)	10.50 (12.30)	35.40 (41.40)
Output of all FRPs through year 1990	Independent	87,000 (102,000)	290,000 (340,000)	5.40 (6.30)	17.30 (20.20)
Output of all FRPs through year 1995	Independent	170,000 (200,000)	590,000 (690,000)	5.30 (6.20)	17.40 (20.30)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 25%.

Only one postulated fuel residue storage accident was identified as causing a radionuclide release, the dropping and breaching of a filled canister. Such an accident was estimated to result in the atmospheric release of about 4 Ci (predominantly activation products).

The low plutonium content and the compact metallic form of the waste and the weight and high surface dose rate of the waste containers make fuel residue in interim storage an unattractive and unavailable target for either theft or sabotage. No special physical protection requirements were identified.

1.3.2.4 Interim Storage of Non-High-Level Solid Waste

Solid wastes that result from the treatment of failed equipment and noncombustible trash, the treatment of combustible and compactable wastes, and the treatment of concentrated liquids, wet wastes, and particulate solids are stored in 55-gal drums, in 1.2 x 1.8 x 1.8 m steel boxes, or in canisters. The canisters are similar to those used for the storage of the fuel residue, and they are stored in the fuel residue storage facility described earlier. Other facilities would be used for storage of the waste in the drums or boxes.

The reference system includes storage of two different fractions, low-level TRU wastes (TRU-LLW) that are handled by direct methods such as shielded fork lift trucks, and intermediate-level TRU wastes (TRU-ILW) that are handled remotely. The TRU-LLW packages have surface radiation levels of <0.2 R/hr. The reference system is based on indoor storage of TRU-ILW and outdoor storage of TRU-LLW. Storage facilities may be located either at the FRP or MOX FFP sites or at independent Federal sites.

In the reference concept, TRU-LLW is stored on an outdoor surface and covered with plastic and earth. The other concept for TRU-LLW storage examined in detail is indoor unshielded storage in conventional warehouses constructed of precast concrete. Cost comparisons for these alternatives at the site of generation and at an independent Federal site are given in Table 1.3.6.

TABLE 1.3.6. Estimated Interim Storage Costs for TRU-LLW

Storage Requirement	Location	Capital Cost, $1000s[a]		Levelized Unit Cost, $/kg HM[a,b]	
		Outdoor	Indoor	Outdoor	Indoor
First 5 years output of one FRP	FRP	1,100 (1,300)	1,300 (1,500)	0.35 (0.40)	0.40 (0.50)
First 5 years output of one MOX FFP	MOX FFP	1,000 (1,200)	1,000 (1,200)	0.32 (0.38)	0.32 (0.38)
Output of all FRP's and MOX FFP's through year 1990	Independent	3,900 (4,600)	5,900 (6,900)	0.30 (0.35)	0.40 (0.45)
Output of all FRPs' and MOX FFPs' through year 1995	Independent	5,700 (6,600)	9,700 (11,000)	0.25 (0.30)	0.35 (0.40)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainties in the unit cost estimates are 20-30%.
b. $/kg HM reprocessed. In the MOX FFP case, multiply by five to obtain $/kg HM of MOX fuel fabricated.

A shielded storage building is the reference concept for storing TRU-ILW. The thick cell walls provide the necessary radiation shielding, and the waste packages are unloaded and handled using a remotely controlled crane. The other storage concept for TRU-ILW examined in detail is outdoor storage of unshielded containers in below grade caissons. Cost comparisons for these alternatives are given in Table 1.3.7.

TABLE 1.3.7. Estimated Interim Storage Costs for TRU-ILW

Storage Requirement	Location	Capital Cost, $1000s[a]		Levelized Unit Cost, $/kg HM[a]	
		Outdoor	Indoor	Outdoor	Indoor
First 5 years output of one FRP	FRP	38,000 (45,000)	16,000 (19,000)	7.90 (9.25)	4.40 (5.15)
Output of all FRP's through year 1990	Independent	110,000 (130,000)	47,000 (55,000)	5.10 (6.00)	2.30 (2.70)
Output of all FRP's through year 1995	Independent	210,000 (250,000)	83,000 (97,000)	5.00 (5.85)	2.20 (2.60)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainties in the unit cost estimates are 25-30%.

Seven postulated accidents were examined for these non-high-level solid waste interim storage alternatives. The most severe atmospheric release is postulated to result from breaching of a drum of ILW; the maximum release is estimated to be about 0.01 Ci.

The overall physical security measures required at sites containing non-high-level waste are sufficient to protect the public from misuse of this waste.

1.3.2.5 Interim Storage of PuO_2

This concept could be used if spent fuel is reprocessed for recycle of uranium before a determination on the disposition of the plutonium has been made. In the reference concept, PuO_2 is stored in a vault-type structure that will contain a concrete storage slab in the floor of the vault. Steel-lined holes house containers specifically designed for PuO_2 storage and transport. These containers are pressure vessels made of stainless steel that are designed to hold canisters of PuO_2 and to withstand an internal pressure of 230 psig. The canisters have filtered vents to allow gases generated in the canisters to pass into the pressure vessel. The pressure vessel is equipped with a manual vent to allow periodic controlled release of gas pressure.

Storage facilities for PuO_2 could be located either at the FRP or at an independent site. Two different sized storage modules at the FRP are described, as well as a large module at an independent site. The costs of using these storage facilities to store the plutonium from a 10-year production at the reference FRP and to store all of the plutonium recovered to the year 2000 in the reference system are compared in Table 1.3.8.

The emission of nonvolatile radionuclides from a full 200 MT PuO_2 module is estimated to be about 10^{-7} Ci/yr.

TABLE 1.3.8. Estimated Costs for Interim Storage of PuO_2

Module Capacity, MT PuO_2	Location	Capital Cost, $1000s[a]		Levelized Unit Cost, $/kg HM Processed[a]	
		Accumulate 10 Years	Accumulate to Year 2000	Accumulate 10 Years	Accumulate to Year 2000
30	FRP	240,000 (280,000)	--	28.80 (33.70)	--
200	FRP	210,000 (250,000)	--	39.70 (46.40)	--
200	Independent	--	900,000 (1,050,000)	--	19.20 (22.50)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainties in the unit cost estimates are 20-30%.

Six postulated accidents were examined for the plutonium oxide storage concept. The most severe radionuclide release is postulated to result from a criticality incident involving 10^{19} fissions; such an incident would release 10^{5} Ci of volatile radionuclides as well as 2×10^{-11} kg of PuO_2 (2×10^{-7} Ci of actinides).

The physical protection and safeguards requirements for a plutonium oxide storage facility would be very extensive. An engineered system incorporating intrusion resistant structures and remote control techniques for normal handling and storage operations would likely be used. Furthermore, movements would be monitored by radiation and metal detectors and alarms and by direct searches at points of exit.

1.4 WASTE TRANSPORTATION TECHNOLOGY

1.4 WASTE TRANSPORTATION TECHNOLOGY

Section 6 analyzes transportation requirements for post-fission TRU wastes that result from the commercial LWR fuel cycle. The materials requiring transportation depend on the fuel cycle. As in the case of interim storage technology (Section 1.3), the transportation technologies are discussed in this summary as they are combined in waste management system applications rather than in the order in which they are described in Section 6. The contents of Section 6 and location where each component is discussed in this summary are as follows:

Report Section	Section 6 Topics	Summary Section
6.1	Background information	1.4.1
6.2	Spent fuel	1.4.2
6.3	Solidified high-level waste	1.4.3.1
6.4	Fuel residue	1.4.3.2
6.5	Plutonium	1.4.3.4
6.6	Non-high-level solid waste	1.4.3.3

1.4.1 Background

All shipments containing nonexempt quantities of radioactive material are regulated by the U.S. Department of Transportation (DOT) and the Nuclear Regulatory Commission (NRC). Exemption from the regulations is based on quantity of material and several other conditions. The regulations prescribe shipping container requirements, limitations on package contents, and packaging and handling procedures. Information is summarized on:

- classification of radioactive materials for shipment
- low specific-activity material
- type A packaging standards
- type B packaging standards
- nuclear criticality safety
- radiation dose rate limitations
- surface contamination levels
- external temperature
- shipment of radioactive materials with other hazardous materials.

Accident experience with radioactive material shipments in routine commerce shows that about one package of radioactive material in 10,000 is involved in some kind of transport accident. However, the incidence of material release is low. Most releases have involved low specific-activity material or type A packaging. Type B packaging, which is designed to withstand severe accident situations, has an excellent record of package integrity.

1.4.2 Waste Transportation in the Once-Through Cycle

The waste transportation activities in the reference once-through system include 1) shipment of the unpackaged fuel from the nuclear power plant to a modified ISFSF colocated with a packaging

facility and 2) shipment of the packaged fuel to the final isolation facility. Alternatively packaged fuel might be shipped to a separate storage facility followed later by shipment to final isolation. Another alternative not considered in detail involves colocation of the packaging facilities with the final repository. This would eliminate one shipping step.

These shipments use massive, heavily shielded shipping casks capable of dissipating the heat resulting from the radioactive decay. Casks for the unpackaged fuel are licensed and available for both truck and rail transport; those for truck transport contain 0.4 to 1.4 MTHM (1 to 3 PWR or 2 to 7 BWR assemblies) and range in weight from 22 to 36 MT and in heat removal capacity from 11 to 36 kW. The casks for rail transport contain 3.2 to 4.5 MTHM (7 to 10 PWR or 18 to 24 BWR assemblies) and range in weight (when loaded) from 63 to 88 MT and have a licensed heat removal capacity from 62 to 70 kW.

Shipment of the packaged fuel would necessitate some changes in existing cask designs; mainly a longer shipping cask and modification of racks for holding fuel in the cask cavity. A suitably modified rail cask based on a design licensed for unpackaged fuel could transport 7 PWR or 17 BWR packaged assemblies.

Cost comparisons for the transportation activities of the once-through cycle are contained in Table 1.4.1. Two types of rail transport are considered: regular train transport and special train transport used solely for the shipment of spent fuel.

Fourteen postulated accidents during shipment of fuel were examined. The most severe accident subjects the cask to severe impact and fire; such an accident is estimated to release 10^{-7} of the contained nonvolatile fission products and actinides as well as 3% of the contained krypton and 1% of the contained iodine and tritium.

TABLE 1.4.1. Waste Transportation Costs for the Once-Through Cycle

Waste	Transport Mode	Unit Cost[a,b] $/kg HM
Unpackaged Spent Fuel	Rail (regular)	14 - 20[c] (16 - 23)
	Rail (special)	18 - 26[c] (21 - 30)
	Truck	16 (19)
Packaged Spent Fuel	Rail (regular)	20[d] (23)
	Rail (special)	26[d] (30)

a. Mid-1976 dollars (mid-1978 dollars in parentheses). The uncertainty in the unit cost estimates is 50%.
b. Based on 1000 mile shipping distance.
c. Estimates for two different presently licensed shipping casks.
d. Based on modification of the more cost-effective cask licensed for unpackaged fuel.

Spent fuel is more accessible to theft or sabotage during transport than when at a fixed site; however, even while it is being transported the fuel is an undesirable target. Additional safeguard procedures are available should their implementation be deemed necessary.

1.4.3 Waste Transportation in the Reprocessing Fuel Cycles

For reprocessing fuel cycles, the TRU-waste transportation operations ship the packaged TRU waste from the FRPs and MOX FFPs to interim storage and/or final repository facilities. Shipment of spent fuel to the FRP is also required, but this is not truly a waste transportation operation in the reprocessing fuel cycles; such shipments would be essentially identical to those described as waste transportation in the once-through cycle.

Packaged TRU wastes that must be shipped in the plutonium plus uranium recycle system are the solidified high-level waste, the fuel residue, and the non-high-level solid wastes. The uranium-only recycle systems also ship these wastes; in addition, plutonium oxide must be shipped to storage if it is not incorporated in the SHLW.

The costs of these reprocessing fuel cycle waste transportation activities are listed in Table 1.4.2. Descriptions of transportation requirements for each waste type follow.

TABLE 1.4.2. Waste Transportation Costs for the Reprocessing Fuel Cycles

Waste	Transport Mode	Unit Cost[a,b] $/kg HM
Solidified High-Level Waste	Rail (regular)	2.80 (3.30)
	Rail (special)	4.20 (4.90)
Fuel Residue	Rail (regular)	3.00 (3.50)
	Rail (special)	5.80 (6.80)
Non-High-Level Solid Waste	Truck and Rail[c]	2.45 (2.90)
Plutonium Oxide[d]	Truck	0.70 (0.80)

a. Mid-1976 dollars (mid-1978 dollars in parentheses) per kg of heavy metal processed, assuming the use of the reference waste treatment alternatives described in Section 1.2. Uncertainties range from 15% for non-high-level waste to 50% for SHLW.
b. Based on 1500-mile shipping distance
c. Failed equipment in canisters similar to fuel residue canisters is shipped by rail.
d. Only in the uranium-only recycle system in which plutonium is stored separately.

1.4.3.1 Transportation of Solidified High-Level Waste

Solidified high-level waste is expected to be shipped in casks that resemble those currently available for shipment of spent fuel. Waste shipments could be by rail or truck; rail shipment is the reference transport system. The reference rail cask weighs about 100 MT, will dissipate

up to 50 kW of internally generated heat, and will hold nine high-level waste containers 30 cm in dia and 3 m in length or 36 containers 15 cm in dia and 3 m long. A special rail car and mounting system is used to transport the cask; the car and mounting equipment weigh about 50 MT.

Five postulated accidents were examined for shipment of solidified high-level waste. The most severe accident involves the combination of severe impact and fire. Such an accident is estimated to release 5×10^{-6} of the contained radionuclides if the waste product is calcine and 5×10^{-8} of the contained radionuclides if the waste product is glass.

The physical protection and safeguard requirements for the transportation of solidified high-level waste would be no greater than those for the transportation of spent fuel.

1.4.3.2 Transportation of Fuel Residue

Fuel residue is also expected to be shipped in casks that resemble those currently available for shipment of spent fuel. However, casks for this purpose could be simpler in design because neutron shielding would not be required and heat removal requirements would be much reduced. Fuel residue shipments could be by rail or truck. Rail shipment is the reference transport system. The reference rail cask is assumed to weigh about 65 MT and is capable of transporting three reference waste canisters.

Four postulated accidents were examined for shipment of fuel residue. The most severe accident involves the combination of severe impact and fire. Such an accident is estimated to release 10^{-6} of the contained radionuclides as respirable particles.

No additional physical protection and safeguards measures would be necessary for fuel residue transport beyond those provided for safety protection.

1.4.3.3 Transportation of Non-High-Level Solid Waste

The steel drums and boxes containing the non-high-level solid waste are presumed to be contained in suitable overpacks during shipment by truck or rail. These overpacks may be shielded or unshielded, depending on the radiation dose at the surface of the disposable container. Drums and boxes which do not require shielding are assumed to be transported in a Super Tiger®, which is a double-walled steel box with a fire-resistant filler. The Super Tiger® weighs 6,800 kg, and the maximum payload is 13,600 kg. A total of 36 drums (55-gal) can be accommodated. A shielded van that meets required package standards or a Super Tiger®-type overpack that incorporates some shielding would have to be constructed for shipping containers with surface dose rates in the 1 R/hr range. Truck casks are available that incorporate several inches of lead or other shielding material and meet required standards. Two such casks are considered in the reference system, 1) a cask with 5 cm lead plus 2 cm steel shielding (suitable for 1 to 10 R/hr drums) and a capacity of fourteen 55-gal drums and 2) a cask with 10 cm lead plus 2.5 cm steel shielding (suitable for >10 R/hr drums) and a capacity of six 55-gal drums.

Four postulated accidents were examined for shipment of non-high-level solid waste. The most severe accident involves the combination of severe impact and fire; such an accident is estimated to release 10^{-5} of the contained radionuclides as respirable particles.

No special safeguarding during shipment of these wastes is required.

1.4.3.4 Transportation of Plutonium

In the uranium-only recycle mode in which plutonium is stored, plutonium dioxide is assumed to be the plutonium compound transported. The reference container for the shipment of PuO_2 is an overpack which contains the pressure vessel in which the PuO_2 is to be stored. The overpack contains a neutron-absorbing solid material and lead to provide gamma shielding. The total weight of the container, which holds 32 kg PuO_2, is about 1640 kg. Truck shipment is the reference transport system, and 10 containers will be transported per shipment.

Because of the stringent controls on packaging and shipment of plutonium oxide, no accidental release of material is postulated for such shipments.

Plutonium oxide is an attractive target for theft, and it would be a more accessible target during transport than at other stages of the fuel cycle. Thus, special regulations have been specified to protect the plutonium from theft or sabotage during shipment.

1.5 FINAL ISOLATION TECHNOLOGY

1.5 FINAL ISOLATION TECHNOLOGY

The objective for the final isolation of radioactive waste is to prevent it from becoming a threat to human health and safety during its hazardous lifetime. Isolation of radioactive wastes in deep geologic repositories is considered an effective means of permanent waste isolation within the scope of available technology. Section 7 describes the geologic considerations essential for repository selection, the nature of geologic formations that are potential repository media, the thermal criteria for waste placement in geologic repositories and conceptual repositories in four different geologic media. Alternatives to geologic isolation are discussed extensively in DOE/EIS-0046-D[1] and are not described in this report.

The contents of Section 7 and the locations where these topics are discussed in this summary are as follows:

Report Section	Section 7 Topics	Summary Section
7.1	Background information on final isolation	1.5.1
7.2	Potential geologic formations	1.5.2
7.3	Geologic repository thermal criteria	1.5.3
7.4	Geologic repositories for the once-through fuel cycle	1.5.4
7.5	Geologic repositories for the reprocessing fuel cycle	1.5.5

1.5.1 Background

Since its earliest days, the nuclear power program has included the study of techniques for the permanent isolation of radioactive wastes. Included in these studies has been the development of technology for converting liquid or soluble wastes to leach-resistant solids and final isolation of the solidified wastes. Solidification not only decreases the potential for an accidental release to the environment, but permits safe shipment to a final isolation site, and generally reduces the volumes of waste to be stored.

The concept of disposing of solidified radioactive wastes in cavities mined in salt beds or domes was suggested in the late 1950s. With demonstration of solidification processes, the feasibility of storing solidified radioactive waste in buried salt was explored in an experiment known as Project Salt Vault[2] conducted in an unused salt mine near Lyons, Kansas, between 1963 and 1968. In this experiment, containers of highly radioactive test reactor fuel, used to simulate the radioactive and thermal properties of solidified waste, were buried (and later retrieved) and the effects on the salt were measured.

Based on the results of the waste solidification program and Project Salt Vault, the AEC proposed use of the abandoned salt mine at Lyons, Kansas, as a pilot facility and an initial

Federal repository for the disposal of commercial high-level wastes, subject to the satisfactory completion of certain additional tests and studies. However, when significant delays became evident in the resolution of site-specific questions on possible routes of entry of water into the mine, the AEC decided to terminate further work at Lyons, and the site was returned to its owners contamination-free.

After the work at Lyons was stopped, studies were expanded to evaluate the suitability of all potential geologic formations and rock types in the continental United States for waste repositories. Geologic formations under investigation included: 1) rock salt, 2) crystalline rocks, 3) argillaceous rocks, 4) carbonate rocks, and 5) volcanic rocks.

Currently, the Office of Nuclear Waste Isolation (ONWI) operated by Battelle Memorial Institute for DOE is conducting broadly-based studies including thermal analysis, rock mechanics, rock-waste interactions, waste migration, risk analysis, etc., on potential repository media, as well as repository design and site selection studies. ONWI is currently engaged in investigation of sites for a repository in a salt formation with plans for investigation of other geologic media.

Ongoing studies in the Columbia River Flood Basalt are being conducted by Rockwell Hanford Operations while similar studies of the Nevada Test Site shales and granites are being conducted by the DOE Nevada Operations Office.

1.5.2 Potential Geologic Formations

The radioactive waste isolation goal may be met in different manners at different sites because of the inherent variability of geologic environments. Accordingly, each prospective repository location must be carefully studied and evaluated to identify the unique geologic features of that specific site. The suitability of a site for waste isolation can be determined only after these characteristics are adequately understood. General requirements for a repository site are classified by the following site characteristics:

- depth of isolation level
- properties and dimensions of host rock
- hydrogeology
- tectonic stability, faulting and seismicity
- relationship to natural resources
- presence of multiple geologic barriers to prevent release of radioactivity.

Four potential repository media, salt, granite, shale, and basalt, are described. Candidate media, however, are not limited to these four types. Other media such as, for example, carbonate rocks may possess the necessary attributes to qualify for selection as repository sites.

Based on typical geologic formations found throughout the U.S., generic site descriptions have been developed for each of the candidate media. These site descriptions include a generic stratigraphic section, hydrologic parameters, mechanical and thermal properties, and an estimation of effects of environmental changes (time, temperature, pressure, moisture, etc.) on the rock formations. These generic site descriptions form the basis for the conceptual repository designs.

1.5.3 Geologic Repository Thermal Criteria

Heat from wastes emplaced in the four geologic media will have definite impacts on:

- the integrity and recoverability of the waste canisters
- mined room and pillar stability
- integrity of the waste matrix over long periods of time
- the integrity of the host rock and the surrounding rock units
- overlying aquifers and bouyancy effects on groundwater flow
- long term uplift and subsidence of overlying rocks.

To assure that the impact of the heat on these factors will not be detrimental to waste isolation objectives, a systematic determination of the repository design thermal loads requires:

- establishment of physical limits for factors affected by heat
- determination of acceptable thermal loads that will not bring about conditions beyond the assigned limits
- development of repository design thermal loads, taking into account safety, engineering, and operational requirements.

Preliminary estimates of acceptable physical limits for factors affected by heat are summarized in Table 1.5.1.

TABLE 1.5.1. Thermal and Thermomechanical Limits for Conceptual Design Studies

Event	Limits
Far Field Considerations	
Maximum uplift over repository	1.2 to 1.5 m[5]
Temperature rise at surface	<0.5°C[6]
Temperature rise in aquifers	<6°C[6]
Near-Field Considerations	
Room closure during ready retrievability period - salt	<10-15% of original room opening[5]
Room stability - granite, basalt rock strength-to-stress ratio	>2 within 1.5 m of openings[7]
Room stability - shale with continuous support rock strength-to-stress ratio	>1 within 1.5 m of openings[7]
Pillar stability - non-salt strength-to-stress ratio	>2 across mid-height of pillar[7]
Very-Near-Field Considerations	
Maximum HLW temperature	
• Glass	500°C[8]
• Calcine	700°C[8]
Maximum spent fuel pin temperature	200°C[8]
Maximum canister temperature	375°C[8]
Maximum salt temperature	250°C[5]
Maximum fracture of non-salt rock	15 cm annulus around canister[5]

The thermal loading limits that were used as a design basis for the conceptual repositories described in this report to meet these physical limits are summarized in Table 1.5.2. The thermal limits for the uranium-only recycle case are identical to those of the once-through cycle if the plutonium is included in the solidified HLW; if the plutonium is stored separately the limits are the same as in the uranium and plutonium recycle case.

TABLE 1.5.2. Conceptual Repository Design Thermal Limits

Repository Medium	Canister Limit kW	Areal Thermal Loading Limits,[a] kW/acre	
		Once-Through Cycle	U & Pu Recycle
Salt	3.2	40	100
Granite	1.7	130	130
Shale	1.2	80	80
Basalt	1.3	130	130

a. These design limits are 2/3 of calculated allowable limits.

1.5.4 Geologic Repositories for the Once-Through Fuel Cycle

Repositories located in salt, granite, shale and basalt formations and serving requirements of the once-through fuel cycle consist of surface facilities that receive the canistered spent fuel assemblies, shafts and hoists that provide access to subsurface facilities, and subsurface facilities that transport and emplace the waste. Repository descriptions are based on conceptual repositories described in References 3 and 4 modified to accommodate the waste forms described in this report. They do not necessarily represent an optimum design but are representative of what could be achieved with current technology. All repository concepts are based on utilizing an 800 ha (2000 acre) underground area. Because of variations in waste emplacement criteria between different media, the capacities of the conceptual repositories are different for each medium.

Waste handling facilities at the repositories contain standard remote handling and hot cell equipment designed to receive canistered spent fuel at the rates shown in Table 1.5.3. Facilities are also provided for receiving and overpacking damaged canisters.

TABLE 1.5.3. Maximum Spent Fuel Receiving Rates

Geologic Medium	Final Year of Repository Operation	Canisters		MTHM	
		PWR	BWR	PWR	BWR
Salt	2000	7,500	11,000	3,500	2,100
Granite	2009	12,000	18,000	5,500	3,400
Shale	2002	9,100	14,000	4,200	2,600
Basalt	2009	12,000	18,000	5,500	3,400

Initially the repositories are operated in a readily retrievable mode. This means that all wastes could be removed from the repositories at about the same rate with about the same effort as for emplacement. Emplacement rooms are not backfilled during this period, waste canisters are placed vertically in steel sleeves that are sealed with a concrete plug in holes or trenches in the floor of the emplacement rooms. The readily retrievable period spans the initial five years of repository operation. This time provides a period for observing waste-rock interactions and repository operations. In addition, all mining and excavation is completed during this five-year period permitting examination of the entire host rock formation within the repository boundaries prior to backfilling of emplacement rooms.

After the readily retrievable period, use of steel sleeves is discontinued and rooms are backfilled as they become filled with canisters. Table 1.5.4 lists the contents of conceptual repositories located in salt, granite, shale, and basalt formations at the end of emplacement. Following completion of waste emplacement operations, corridors will be backfilled, surface facilities decommissioned and shafts backfilled and sealed.

TABLE 1.5.4. Contents of Alternative First Repositories

		PWR		BWR	
	Year	Canisters	MTHM	Canisters	MTHM
Salt	2000	68,200	31,500	104,000	19,600
Granite	2009	162,700	75,100	246,300	46,500
Shale	2002	86,300	39,800	131,000	24,700
Basalt	2009	162,700	75,100	246,300	46,500

Table 1.5.5 compares costs for the repositories in salt, granite, shale, and basalt.

TABLE 1.5.5. Costs for Repositories for the Once-Through Fuel cycle[a]

Geologic Medium	Construction and Mining Cost, 10^6	Total Operating Costs, 10^6	Decommissioning Cost, 10^6	Levelized Unit Cost, $/kg HM[b]
Salt	880 (1,030)	500 (590)	21 (24)	44 (52)
Granite	2,200 (2,600)	2,020 (2,360)	21 (24)	67 (78)
Shale	1,100 (1,300)	690 (810)	21 (24)	49 (57)
Basalt	2,600 (3,000)	2,040 (2,390)	21 (24)	74 (87)

a. Mid-1976 dollars (mid-1978 dollars in parentheses).
b. Estimated uncertainty is 50%.

Although no routine radioactive emissions were identified for the repositories, the integrated annual release from minor accidents is postulated to result in the release of 5 Ci/yr of volatile radionuclides (essentially all ^{85}Kr) but no significant release of non-volatile radionuclides.

Six design basis and eight very low probability non-design basis accidents such as a strike by a meteorite on a major facility, were examined for these isolation technologies. The only accident expected to occur with a significant frequency that results in a release of radionuclides is a minor canister failure caused by rough handling. A radionuclide release of 3 Ci (of ^{85}Kr) is postulated to result from such an accident.

The physical protection and safeguards requirements for the once-through fuel cycle repositories would be comparable to those used at other fuel cycle facilities in which spent fuel is handled. The surface facilities at the repositories would receive the principal emphasis.

1.5.5 Geologic Repositories for the Reprocessing Fuel Cycles

Section 7.5 describes repositories located in salt, granite, shale, and basalt formations operating in the three reprocessing fuel cycles: 1) uranium-only recycle with plutonium in the high-level waste (Cycle 2a), 2) uranium-only recycle with PuO_2 stored for future use or disposal (Cycle 2b), and 3) uranium and plutonium recycle (Cycle 3). A repository operating in support of any of these reprocessing fuel cycles is required to receive high-level waste (HLW), fuel residue waste (FRW), and intermediate- and low-level transuranic (TRU) wastes. The repository designs are based on conceptual repositories described in References 3 and 4 modified to accommodate the treated waste forms described in this report. The designs do not necessarily represent an optimum design but are representative of what could be achieved with current technology. As in the once-through fuel cycle repositories, all repository concepts are based on an 800 ha (2000 acre) underground area. Because of variations in waste emplacement criteria among different media, the capacities of the conceptual repositories are different for each medium.

Waste handling facilities at these repositories include standard remote handling and hot cell equipment designed to receive HLW, FRW, and ILW, and shielded materials handling equipment to receive LLW. These facilities are designed to receive wastes at the rates shown in Table 1.5.6.

Repositories for the reprocessing fuel cycles operate in the same manner and with the same initial readily retrievable period as described for the once-through fuel cycle repository. Table 1.5.7 lists the contents of conceptual alternative first repositories located in salt, granite, shale, and basalt at the end of emplacement operations. As in the case of the once-through fuel cycle repositories, wastes are emplaced in a readily retreivable mode using steel sleeves and concrete plugs for the canistered wastes for the first five years. All mining is completed during this five-year period.

Table 1.5.8 compares the costs for the repositories in salt, granite, shale, and basalt.

No routine radioactive emissions were identified for the repositories. The integrated annual release due to minor accidents for these facilities is estimated to be very low.

TABLE 1.5.6. Maximum Waste Receiving Rates

	Final Year	Number of Containers per Year					
		HLW Canisters[d]	FRW Canisters	ILW Canisters	ILW Drums	LLW Boxes	LLW Drums
Cycle 2a[a]							
Salt	1999	2,300	1,600	240	27,000	170	9,100
Granite	2004	5,100	2,100	310	35,700	220	11,800
Shale	1997	4,600	1,300	200	22,000	140	7,400
Basalt	2002	7,200	1,800	270	30,000	190	10,000
Cycle 2b[b]							
Salt	2005	2,500	2,300	330	38,000	280	13,000
Granite	2004	5,100	2,100	310	36,000	270	12,000
Shale	1997	4,600	1,300	200	22,000	170	7,500
Basalt	2002	7,200	1,800	270	30,500	230	10,300
Cycle 3[c]							
Salt	2003	2,900	2,000	290	33,000	350	23,000
Granite	2004	5,100	2,100	210	36,000	380	24,600
Shale	1997	4,600	1,300	200	22,000	240	15,600
Basalt	2002	7,200	1,800	270	30,500	330	21,400

a. Cycle 2a is uranium-only recycle with plutonium in the HLW.
b. Cycle 2b is uranium-only recycle with PuO_2 stored for future use or disposal.
c. Cycle 3 is uranium and plutonium recycle.
d. HLW canister size depends on thermal limit of media and varies from 15 to 30 cm diameter.

Nine design basis and eight very low probability non-design basis accidents were examined for these repositories when used for reprocessing fuel cycle wastes. Only negligible atmospheric release of radionuclides is postulated to result from accidents that are expected to occur with any significant frequency.

The physical protection and safeguards requirements for the reprocessing fuel cycle repositories would be comparable to those used at the facilities at which the wastes are generated and stored. The surface facilities at the repositories would receive the principal emphasis.

TABLE 1.5.7. Contents of Alternative First Repositories

Fuel Cycle	Waste	Salt		Granite		Shale		Basalt	
		Containers	Equivalent[a] MTHM	Containers	Equivalent MTHM	Containers	Equivalent MTHM	Containers	Equivalent MTHM
2a - Uranium only recycle, Plutonium in HLW									
		(1999)[c]		(2004)		(1997)		(2002)	
	HLW canisters[b]	15,900	39,500	49,200	69,000	37,700	30,500	69,600	56,000
	FRW canisters	15,900	69,000	25,300	108,500	13,100	56,000	21,500	91,500
	ILW canisters	2,370	69,000	3,760	108,500	1,950	56,000	3,190	91,500
	ILW drums	270,000	69,000	427,000	108,500	222,000	56,000	363,000	91,500
	LLW boxes	1,670	69,000	2,650	108,500	1,370	56,000	2,250	91,500
	LLW drums	88,500	69,000	141,000	108,500	72,800	56,000	119,000	91,500
2b - Uranium only recycle, Plutonium stored									
		(2005)		(2004)		(1997)		(2002)	
	HLW canisters[b]	24,600	76,500	48,800	69,000	36,600	30,500	61,700	56,000
	FRW canisters	27,500	118,000	25,300	108,500	13,100	56,000	21,500	91,500
	ILW canisters	4,090	118,000	3,760	108,500	1,950	56,000	3,190	91,500
	ILW drums	469,000	118,000	431,000	108,500	224,000	56,000	367,000	91,500
	LLW boxes	3,450	118,000	3,170	108,500	2,290	56,000	2,700	91,500
	LLW drums	158,000	118,000	145,000	108,500	75,500	56,000	124,000	91,500
3 - Uranium and Plutonium recycle									
		(2003)		(2004)		(1997)		(2002)	
	HLW canisters[b]	25,800	62,170	48,800	69,000	36,600	30,500	63,800	56,000
	FRW canisters	23,400	99,670	25,300	108,500	13,100	56,000	21,500	91,500
	ILW canisters	3,480	99,670	3,760	108,500	1,950	56,000	3,190	91,500
	ILW drums	399,000	99,670	431,000	108,500	224,000	56,000	367,000	91,500
	LLW boxes	4,150	99,670	4,500	108,500	2,290	56,000	3,810	91,500
	LLW drums	264,000	99,670	286,000	108,500	144,000	56,000	242,000	91,500

a. Tonnes of reprocessed heavy metal corresponding to wastes contained in the repository.
b. Equivalent MTHM for HLW does not correspond to the MTHM for other waste types because of 5 year delay for HLW cooling between reprocessing and shipment to the repository.
c. Final year of operation assuming 1985 repository startup.

TABLE 1.5.8. Costs for Repositories for the Reprocessing Fuel Cycles[a]

Fuel Cycle[b]	Geologic Medium	Construction and Mining Cost, 10^6	Total Operating Costs, 10^6	Decommissioning Cost, 10^6	Levelized Unit Cost, $/kg HM[c]
2a	Salt	930 (1100)	710 (830)	24 (29)	48 (56)
	Granite	1700 (2000)	1560 (1880)	24 (28)	65 (76)
	Shale	1100 (1300)	680 (800)	24 (28)	61 (72)
	Basalt	2000 (2300)	1390 (1630)	24 (28)	78 (92)
2b	Salt	1000 (1200)	1090 (1280)	25 (29)	36 (42)
	Granite	1700 (2000)	1610 (1880)	25 (29)	65 (76)
	Shale	1100 (1300)	680 (800)	25 (29)	62 (72)
	Basalt	1900 (2300)	1390 (1630)	25 (29)	77 (90)
3	Salt	1000 (1200)	1030 (1210)	25 (29)	41 (48)
	Granite	1800 (2000)	1660 (1940)	25 (29)	66 (77)
	Shale	1100 (1300)	710 (830)	25 (29)	62 (73)
	Basalt	2000 (2300)	1490 (1740)	25 (29)	80 (93)

a. Mid-1976 dollars (mid-1978 dollars in parentheses).
b. Fuel cycle 2a is uranium-only recycle with plutonium to HLW.
Fuel cycle 2b is uranium-only recycle with plutonium stored for future disposition.
Fuel cycle 3 is uranium and plutonium recycle.
c. Estimated uncertainty is 50%.

REFERENCES FOR SECTION 1.5

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1.6 RETIRED FACILITIES DECOMMISSIONING TECHNOLOGY

1.6 RETIRED FACILITIES DECOMMISSIONING TECHNOLOGY

Reactors and post-fission fuel cycle facilities become contaminated with radioactivity in the course of power production, and in fuel cycle and waste treatment operations. Upon retirement these facilities become components of the fuel cycle wastes. Management of these wastes is commonly termed decommissioning. Various alternatives for decommissioning these retired facilities are described in Section 8. Two basic decommissioning modes are considered. It is assumed in both alternatives that dismantlement will be required:

- Immediate Dismantlement - At shutdown all radioactive contamination above regulatory limits is removed from the facility to an approved disposal site. Depending on further uses of the site, noncontaminated portions of the facility remaining after dismantlement may be demolished and removed.
- Safe Storage with Deferred Dismantlement - At shutdown the facility is prepared to be left in place for an extended time. Continuing surveillance and maintenance are required at the facility. Safe storage methods considered in this report are passive safe storage and hardened safe storage. (These storage methods have also been referred to respectively as protective storage and entombment.) For passive storage, the radioactivity in the facility at shutdown is isolated by installing temporary physical barriers. When the facility's radioactivity levels after passive storage have been sufficiently reduced by decay, the facility would usually be dismantled. For hardened storage, radioactive materials in the facility as shutdown are isolated by installing hardened physical barriers. The facility is maintained in this condition until it is dismantled, or all residual radioactivity has decayed to nonhazardous levels.

Decommissioning activities can be divided into the following general phases:

- The planning and preparation phase is typically carried out during the final 1 to 2 years of facility operation. Decommissioning plans are prepared during this period, and necessary documentation is submitted to NRC for license revisions at facility shutdown.
- The actual decommissioning activities are conducted during the decommissioning operations phase. These activities could include chemical decontamination, mechanical decontamination, equipment deactivation and removal, isolation of contaminated areas if safe storage is to be used and, ultimately, complete dismantlement. The decommissioning techniques used at a particular facility depend on the decommissioning mode selected and the conditions at the facility at shutdown.
- Continuing care activities, when a facility is not immediately dismantled, are carried out at a facility after it is placed in safe storage. These activities assure that the facility remains in a condition that poses minimal risks to the public.
- Final decommissioning activities are required at the end of the continuing care period to permit termination of the facility license. These activities may include performance of radiation surveys to assure that all radioactivity has decayed to levels required by regulation and removal of all residual radioactivity to an approved disposal site.

Two alternative decommissioning modes have been studied for each of the four reference fuel cycle facilities considered in this report. The facility considered in the analysis was assumed to contain operating and waste treatment facilities representative of a complete operating plant.

Nuclear Power Plant. Passive safe storage followed by dismantlement 50 years after reactor shutdown is the reference alternative. The 50-year delay allows decay of most of the ^{60}Co in the facility and reduces occupational radiation exposure during the decommissioning. Immediate dismantlement is also assessed.

Independent Spent Fuel Storage Basin (ISFSB). Immediate dismantlement is the reference decommissioning alternative for this facility. Hardened safe storage is also considered. Surveillance would be required at the facility for 100 years or more after it is placed in hardened safe storage, or until it is dismantled.

Fuel Reprocessing Plant (FRP). The reference decommissioning alternative is passive safe storage followed by dismantlement 30 years after facility shutdown. Hardened safe storage for an extended time (perhaps as long as 1000 years) prior to final decommissioning is also considered.

Mixed Oxide Fuel Fabrication Plant (MOX FFP). Immediate dismantlement is the reference decommissioning alternative for this facility. Hardened safe storage for an extended time prior to final decommissioning is also considered.

Estimates of the costs incurred, the volumes of wastes involved, and the radioactivity contents of the wastes resulting from implementation of the reference decommissioning alternatives are summarized in Table 1.6.1.

TABLE 1.6.1. Reference Fuel Cycle Facilities Decommissioning Alternatives, Costs, Waste Volumes, and Radioactivity

Facility	Operation	Cost, $1000s[a]		Waste Volume, m^3	Radioactivity in Waste, Ci
Nuclear Power Plant	Initial decommissioning	5,900	(6,900)	350	6
	50-yr continuing care	4,100	(4,800)	15	2×10^{-3}
	Deferred dismantlement	15,000	(18,000)	7900	7×10^{4}
ISFSB	Immediate dismantlement	5,000	(5,800)	4700	7×10^{2}
FRP	Initial decommissioning	6,000	(7,000)	830	3×10^{3}
	30-yr continuing care	5,000	(5,800)	10	1
	Deferred dismantlement	22,000	(26,000)	8800	5×10^{3}
MOX FFP	Immediate dismantlement	9,000	(10,000)	2800	2×10^{5}

a. Mid-1976 dollars (mid-1978 dollars in parentheses).

Accidents that might occur during decommissioning were concluded to be much lower in both frequency and overall consequence than accidents that would occur during normal plant operations.

The decommissioning plan prepared and approved prior to decommissioning would detail the physical security and safeguards features to be implemented (as needed) during the decommissioning operations. The features already in place at the operating plants would likely suffice during the decommissioning operations as well.

1.7 THORIUM FUEL CYCLE CONSIDERATIONS

1.7 THORIUM FUEL CYCLE CONSIDERATIONS

Section 9 contains a brief discussion of thorium fuel cycle considerations. The high-temperature gas-cooled reactor fuel cycle and the light-water breeder reactor fuel cycle are described and the wastes from these cycles are compared to the uranium fuel cycle discussed in this report. The long term waste characteristics of the thorium cycle appear, in general, to be similar to those from the uranium fuel cycle. A precise comparison of these characteristics cannot be made without defining specific details of all processing and neutron exposure histories.

1.8 WASTE MANAGEMENT SYSTEM

1.8 WASTE MANAGEMENT SYSTEM

Analysis of the complete waste management system was developed to assess the total impact of managing radioactive wastes generated over the entire lifetime of a nuclear power system. The analysis considered treatment and disposal of all post-fission TRU (including spent fuel and HLW), gaseous, airborne, and decommissioning wastes. Each radioactive waste stream is tracked each year from origin through treatment, storage, transport, and accumulation in a geologic repository.

The waste managment system analysis methodology and results are presented in Section 10. The contents of section 10 and the locations where these topics are discussed in the Summary are as follows:

Report Section	Section 10 Topics	Summary Section
10.1	System simulation	1.8.1
10.2	Input data for the system	1.8.2
10.3	Total system waste quantities and radioactivity	1.8.3
10.4	Comparison of subsystem costs of waste management alternatives	1.8.4
10.5	Total system costs	1.8.5
10.6	Total system plutonium disposition	1.8.6

The reference nuclear power system is based on 400 GWe of nuclear power installed in the year 2000 and produces approximately 10,000 GWe-yr of electric energy. An alternative low-growth projection based on 255 GWe in the year 2000 is also considered but for fewer cases. This system produces approximately 6400 GWe-yr of electric energy. The lifetime of the system extends from 1975 through 2075 and in addition the decay of radioactivity in the final repository is followed over a 1 million-year period. The fuel cycle cases considered include:

Reference Projection Cases

Case 1 Once-through cycle - 1985 repository
Case 2a Uranium-only recycle with plutonium in the SHLW - 1985 repository
Case 2b Uranium-only recycle with plutonium stored as PuO_2 - 1985 repository
Case 3a Uranium and plutonium recycle - 1985 repository
Case 3b Uranium and plutonium recycle with repository delayed to year 2000
Case 4a* Once-through cycle with repository delayed to year 2000
Case 4a* Deferred decision to year 2000 for once-through fuel cycle
Case 4b Deferred decision to year 2000 for uranium and plutonium recycle

* Case 4a serves a dual purpose.

Low Growth Projection Cases

Case 1 Once-through cycle - 1985 repository

Case 3a Uranium and plutonium recycle - 1985 repository

Results of calculations reported here for each fuel cycle case include:

- quantities of all treated wastes, radioactivity, and heat generation rates for each treated waste component
- radioactivity and heat accumulations in repositories by nuclide, waste type, and totals
- costs for each major waste management function
- unit power costs and unit fuel costs for the total waste management system.

Summary results are presented in the text of Section 10 with detailed results tabulated in the appendices. Whereas costs in previous sections of this report have been presented as 1976 constant dollar costs, all costs in this waste management system section have been converted to 1978 constant dollars.

Radioactivity releases from each waste management operation and resulting population radiation doses were also calculated for the entire system for each fuel cycle case. These results are presented in DOE/ET-0029.

1.8.1

A computer model consisting of 4 major modules was developed for the system simulation. The 4 modules are:

- ORIGEN, which carries out fuel irradiation calculations
- ENFORM, which calculates fuel cycle logistics
- WASTRAC, which calculates waste processing logistics
- IMPACT, which calculates annual and cumulative impacts and system costs

1.8.2 Input Data for the System

Input data for the waste management system calculations were derived from the waste management technology analyses in previous sections of the report. Essential data for these calculations has been consolidated and classified in four series of tables (identified as Data Sheets) in Appendix 10A. These four series of tables include:

- Waste Treatment Data Sheets
- Waste Storage Data Sheets
- Decommissioning Data Sheets
- Unit Cost Data Sheets

1.8.3 Total System Waste Quantities in Radioactivity

Detailed descriptions of the spent fuel logistics, i.e., movement and inventory accumulation, the total quantities of TRU wastes processed in the entire waste management system, and the average radioactivity and heat generation rates in the waste containers are detailed in a series

of tables in Appendix 10B. In addition, the total accumulation of radioactivity in all repositories is described in other tables in Appendix 10B and its decay is tracked over a one million-year period. The total fission and activation product radioactivity accumulations in all repositories and the total actinide radioactivity accumulation in all repositories are summarized in Tables 1.8.1 and 1.8.2.

TABLE 1.8.1. Total Fission and Activation Product Radioactivity Accumulations in All Repositories, Ci

Case	Year		Geologic Time (Years Beyond 1975)			
	2000	2050	1,000	10,000	100,000	1,000,000
1	1.24×10^{10}	5.04×10^{10}	8.24×10^{6}	7.58×10^{6}	5.51×10^{6}	1.09×10^{6}
2A	1.04×10^{10}	4.95×10^{10}	8.44×10^{6}	7.65×10^{6}	5.52×10^{6}	1.13×10^{6}
2B	1.04×10^{10}	4.95×10^{10}	8.44×10^{6}	7.65×10^{6}	5.52×10^{6}	1.13×10^{6}
3A	1.03×10^{10}	4.74×10^{10}	8.29×10^{6}	7.53×10^{6}	5.46×10^{6}	1.11×10^{6}
3B	1.63×10^{9}	4.74×10^{10}	8.29×10^{6}	7.53×10^{6}	5.46×10^{6}	1.11×10^{6}
4A	1.79×10^{9}	5.04×10^{10}	8.24×10^{6}	7.58×10^{6}	5.51×10^{6}	1.09×10^{6}
4B	---	4.85×10^{10}	8.41×10^{6}	7.63×10^{6}	5.51×10^{6}	1.12×10^{6}
1LG	1.02×10^{10}	3.07×10^{10}	5.25×10^{6}	4.83×10^{6}	3.51×10^{6}	6.93×10^{5}
3LG	8.32×10^{9}	2.90×10^{10}	5.29×10^{6}	4.80×10^{6}	3.48×10^{6}	7.06×10^{5}

Note: Case 1 = Once-Through
Case 2A = U-Only Recycle, Pu in HLW
Case 2B = U-Only Recycle, Pu Stored
Case 3A = U & Pu Recycle
Case 3B = U & Pu Recycle, Delayed Repository
Case 4A = Once-Through, Delayed Repository or Deferred Decision
Case 4B = U & Pu Recycle, Deferred Decision
Case 1LG = Low-Growth, Once-Through
Case 3LG = Low-Growth, U & Pu Recycle

Except for the low-growth effect, the fission and activation product accumulations are quite similar in all cases; however, the actinide accumulations have significant differences resulting from the disposition of plutonium in the fuel cycle.

1.8.4 Comparison of Subsystem Costs of Waste Management Alternatives

The costs of waste management alternatives are compared on the basis of complete subsystem costs with final isolation in a salt formation repository. Subsystem costs are defined as all the costs incurred in the management of a specific type of waste from treatment to final disposal. Subsystem unit costs are developed for management of spent fuel as a waste in the once-through cycle, (Cases 1 and 4A) and for management of TRU wastes produced in the plutonium and uranium recycle fuel cycle (Case 3). The subsystem costs take into account differences in treatment costs and the effect that differing waste volumes resulting from treatment have on transportation costs, interim storage costs, and final repository costs. In the once-through cycle the subsystem costs are identical to the system costs because spent fuel is the only waste type requiring geologic isolation.

TABLE 1.8.2. Total Actinide Radioactivity Accumulations in all Repositories, Ci

Case	Year		Geologic Time (Years Beyond 1975)			
	2000	2050	1,000	10,000	100,000	1,000,000
1	3.27×10^9	9.90×10^9	5.97×10^8	1.52×10^8	1.35×10^7	5.99×10^6
2A	2.79×10^9	1.05×10^{10}	5.97×10^8	1.53×10^8	1.32×10^7	5.24×10^6
2B	9.73×10^7	4.34×10^8	6.33×10^7	7.11×10^8	1.63×10^6	2.42×10^6
3A	2.07×10^8	2.30×10^9	1.99×10^8	3.49×10^7	2.58×10^6	2.79×10^6
3B	4.21×10^7	2.30×10^9	1.99×10^8	3.49×10^7	2.58×10^6	2.79×10^6
4A	4.63×10^8	9.91×10^9	5.97×10^8	1.52×10^8	1.35×10^7	5.99×10^6
4B	---	1.81×10^9	2.68×10^8	1.38×10^7	2.21×10^6	3.16×10^6
1LG	2.68×10^9	5.94×10^9	3.80×10^8	9.68×10^7	8.60×10^6	3.82×10^6
3LG	1.59×10^8	1.40×10^9	1.34×10^8	2.15×10^7	1.66×10^6	1.84×10^6

Note: Case 1 = Once-Through
Case 2A = U-Only Recycle, Pu in HLW
Case 2B = U-Only Recycle, Pu Stored
Case 3A = U & Pu Recycle
Case 3B = U & Pu Recycle, Delayed Repository
Case 4A = Once-Through, Delayed Repository or Deferred Decision
Case 4B = U & Pu Recycle, Deferred Decision
Case 1LG = Low-Growth, Once-Through
Case 3LG = Low-Growth, U & Pu Recycle

Table 1.8.3 summarizes average and levelized unit costs over the entire system life for the once-through fuel cycle alternatives. The unit costs are shown as equivalent power costs in mills/kW-hr and unit fuel costs in dollars/kgHM. The table shows results for both early repository availability (Case 1) and delayed repository availability (Case 4A). Very little difference in the total unit fuel cycle costs is indicated in these two fuel cycle alternatives. A significant difference in the fuel cycle costs result, however, from federal or private ownership of the fuel packaging and interim storage facilities. This is because of the higher return on investment required for private industry facilities and federal taxes paid in the private ownership case.

TABLE 1.8.3. Average and Levelized Unit Cost Over System Life for Once-Through Fuel Cycle Alternatives

	Unit Fuel Cost, $/kgHM		Unit Power Cost, mills/kWh	
	0% Discount Rate	7% Discount Rate	0% Discount Rate	7% Discount Rate
Early Repository (Case 1)				
Federal Alternative	170	117	0.73	0.45
Private Alternative	206	143	0.88	0.55
Delayed Repository (Case 4A)				
Federal Alternative	172	110	0.74	0.42
Private Alternative	227	149	0.97	0.57
Maximum Variation for Extended Storage Alternatives	+2	+4	+0.01	+0.01

The average unit cost and levelized unit costs employing a 7% discount rate for major subsystem alternatives for the uranium and plutonium fuel cycle alternative are shown in Table 1.8.4. Comparisons are presented for four major subsystem combinations. Results show:

- selection of either HLW treatment alternative does not affect power costs by more than 0.01 mills/kW-hr
- the cost variation between the fuel residue treatment alternative does not affect power costs by more than 0.02 mills/kW-hr.
- the largest impact of waste management treatment alternative selection is in the management of low-level and intermediate TRU wastes where differences as large as 0.15 mills/kW-hr are observed.
- selection of gaseous waste treatment alternatives can affect power costs by up to 0.1 mills/kW-hr.

TABLE 1.8.4. Average Subsystem Unit Cost of Waste Management Alternatives in the Uranium and Plutonium Recycle Fuel Cycle

	Unit Fuel Cost, $/kg Discharged		Unit Power Cost, mills/kW-hr	
	0% Discount Rate	7% Discount Rate	0% Discount Rate	7% Discount Rate
High-Level Waste				
Vitrified (Reference	59	38	.25	.14
Calcified	61	40	.26	.15
Fuel Residue				
Minimum Treatment (Reference)	11	9	.05	.04
Compaction	8	7	.03	.03
Melting	8	6	.03	.02
Intermediate and Low-Level TRU Wastes				
Incineration-cementation (Reference)	38	31	.16	.12
Incineration-bitumenization	33	27	.14	.10
Minimum treatment-cementation	72	57	.31	.21
Minimum treatment-bitumenization	68	55	.29	.21
Gaseous Waste Treatment				
APS + I, Ru, Kr, C-14 Removal (Reference)	28	23	.12	.09
APS + I, Ru, Kr removal	28	22	.12	.09
APS + I, Ru, C-14 removal	9	7	.04	.03
APS + I, Ru, removal	10	8	.04	.03
APS only	2	1	.01	.01

1.8.5 Total System Costs

Total system costs for the entire waste management system for the 9 fuel cycle cases were developed based on the use of the reference waste management alternatives. All spent fuel,

high-level and other TRU waste management costs including treatment, interim storage, transportation, and disposal in geologic repositories are included. Costs for decommissioning the ISFSFs, FRP, and MOX FFP and the management of decommissioning TRU wastes in the reprocessing cases are also included. Nuclear power plant decommissioning costs are not included. In the fuel reprocessing cases, storage of spent fuel prior to reprocessing, except for the initial 6 months at the power plant basins, and shipment of unpackaged spent fuel to reprocessing are included as waste management costs, although these are not necessarily exclusively waste management costs.

The total system costs for the 9 fuel cycle cases are detailed in a series of 195 tables in Appendix 10E. These tables are grouped by fuel cycle case and show waste management costs at 5-year intervals over the entire life of the system. There are 20 tables for each once-through fuel cycle case and 28 tables for each reprocessing fuel cycle case. Costs were developed using 0%, 7%, and 10% discount rates and for four different repository media; salt, granite, shale and basalt. Results of the calculations are summarized in Tables 1.8.5 through 1.8.7. Table 1.8.5 shows the total unit power costs for TRU waste management, including spent fuel handling and

TABLE 1.8.5. Total Unit Power Cost for TRU-Waste Management Including Spent Fuel Handling and Storage, Mills/kWh

	0% Discount Rate				7% Discount Rate				10% Discount Rate			
Case	Salt	Granite	Shale	Basalt	Salt	Granite	Shale	Basalt	Salt	Granite	Shale	Basalt
1	.73	.84	.75	.88	.45	.51	.46	.53	.38	.42	.39	.44
2A	.76	.96	.88	.97	.52	.61	.59	.65	.45	.52	.51	.56
2B	.81	.97	.99	1.07	.56	.66	.67	.72	.49	.57	.58	.62
3A	.74	.97	.89	.98	.50	.61	.60	.65	.44	.52	.52	.55
3B	.76	.98	.89	.98	.49	.57	.55	.58	.42	.46	.45	.48
4A	.74	.85	.76	.89	.42	.47	.43	.48	.33	.36	.34	.38
4B	1.04	1.19	1.14	1.16	.40	.42	.41	.41	.30	.31	.30	.31
ILG	.73				.45				.37			
3LG	.74				.49				.43			

TABLE 1.8.6. Total Unit Fuel Cost for TRU-Waste Management Including Spent Fuel Handling and Storage, $/kg

	0% Discount Rate				7% Discount Rate				10% Discount Rate			
Case	Salt	Granite	Shale	Basalt	Salt	Granite	Shale	Basalt	Salt	Granite	Shale	Basalt
1	170	195	175	205	117	133	120	139	100	113	102	118
2A	178	224	206	227	136	160	155	169	120	139	136	148
2B	188	225	230	251	145	172	174	187	130	152	154	164
3A	172	226	208	229	131	160	156	169	116	139	137	147
3B	177	229	209	229	128	149	143	152	111	123	120	127
4A	172	198	177	208	110	122	112	126	89	97	91	100
4B	243	270	266	279	104	108	107	109	79	81	81	82
ILG	170				116				98			
3LG	173				129				114			

TABLE 1.8.7. Present Worth System Waste Management Costs Including Spent Fuel Handling and Storage, $ Billions

	0% Discount Rate				7% Discount Rate				10% Discount Rate			
Case	Salt	Granite	Shale	Basalt	Salt	Granite	Shale	Basalt	Salt	Granite	Shale	Basalt
1	64.3	74.1	66.3	77.7	7.4	8.4	7.6	8.7	3.8	4.3	3.9	4.5
2A	66.1	82.4	76.2	83.7	8.3	9.7	9.4	10.2	4.5	5.2	5.1	5.5
2B	69.9	83.1	85.3	92.5	8.9	10.5	10.7	11.4	4.9	5.7	5.7	6.1
3A	65.2	85.9	79.2	86.9	8.2	10.1	9.8	10.6	4.5	5.3	5.3	5.6
3B	67.2	87.0	79.3	87.1	8.0	9.3	8.9	9.6	4.2	4.7	4.6	4.9
4A	65.4	75.2	67.3	78.7	6.9	7.6	7.0	7.9	3.4	3.7	3.5	3.8
4B	92.3	102.6	101.0	106.0	6.5	6.8	6.7	6.9	3.0	3.1	3.1	3.1
ILG	41.1				5.4				2.9			
3LG	4.9				5.9				3.4			

storage, in mills/kW-hr for each of the 9 fuel cycle cases and the four repository media. Table 1.8.6 has a similar format showing total unit fuel costs in $/kgHM. Table 1.8.7 also has the same format and shows total present worth waste management cost in billions of dollars.

The results of the cost analysis indicate that the total system unit costs for waste management are 1) relatively insensitive to fuel cycle selection and 2) likely to fall within the range of 0.5 to 1.0 mill/kW-hr in equivalent 1978 constant dollar power costs.

1.8.6 Total System Plutonium Disposition

An important consideration for different fuel cycle options is the disposition of plutonium. In the once-through cycle and the uranium-only recycle option, a total of 3090 and 3170 MT of plutonium respectively are produced. This plutonium is sent to repository in the once-through cycle and in the uranium-only recycle case it is separated but is either added to the solidified high-level waste for disposal or stored for later use.

In the uranium and plutonium recycle option a total 5700 MT of plutonium is recovered, 4400 tons are recycled leaving 1300 MT that are not recycled. Of this 1300 MT, however, 600 tons were not recycled as a calculation simplification and because of the low fissile plutonium content (this is plutonium recovered from the third recycle and is 47% fissile), leaving 700 MT available for recycle but not used in this system. It is assumed that other power plants not in the reference system would be permitted to use this 700 MT of plutonium. Alternatively, the plutonium that is unused could be reduced to zero in the plutonium recycle fuel cycle case by discontinuing recycle prior to the end of system life and converting to a once-through fuel cycle for the last years of the system life.

2.0 INTRODUCTION

2.0 INTRODUCTION

This report, DOE/ET-0028, was prepared in support of the Environmental Impact Statement on Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, (referred to here as the Commercial Waste Management Statement (CWMS)). DOE/ET-0028 describes the technology basis for the predisposal and geologic repository facilities for the CWMS, while the other companion report in this series, Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, details the environmental effects analyses. These reports have been prepared separately from the CWMS to reduce its length and complexity. This approach responds to the President's executive order 11991 of May 24, 1977 that instructs the Council on Environment Quality to issue regulations to Federal agencies requiring impact statements to be "concise, clear, to-the-point, and supported by evidence that agencies have made the necessary environmental analyses." An earlier report, Alternatives for Managing Wastes in Reactors and Post-Fission Operations in the LWR Fuel Cycle (ERDA-76-43), broadly reviewed the status of radioactive waste management and provided the basis for selection of waste management alternatives examined in detail in this report.

To develop the technological basis for the CWMS that is presented in this report, experts in various areas of waste management technology from twelve DOE contractors and laboratories were appointed as Task Leaders. The laboratories and organizations participating in these technology tasks included:

Argonne National Laboratory (ANL)

Allied Chemical Corporation, Idaho (ACC)

Atlantic Richfield Hanford Company (ARHCO, which has been reorganized as Rockwell Hanford Operations)

Hanford Engineering Development Laboratory (HEDL)

Los Alamos Scientific Laboratory (LASL)

Oak Ridge National Laboratory (ORNL)

Office of Waste Isolation, Oak Ridge (OWI)

Savannah River Laboratory (SRL)

Sandia Laboratory (SL)

Pacific Northwest Laboratory (PNL)

EG&G, Idaho

Office of Nuclear Waste Isolation (ONWI)

Altogether, 24 different task groups contributed to this report. In addition, an architect-engineering (A-E) firm, Bechtel, Inc., was contracted to prepare conceptual designs based on facility descriptions developed by the task groups and to prepare facility capital cost estimates and construction requirements and impacts. Other subcontracted technical support was provided by Science Applications Inc., Parsons, Brinkerhoff, Quade and Douglas, Inc., and Dames and Moore. Staff at the Pacific Northwest Laboratory (PNL) served as task coordinators in compiling and coordinating the information developed by the task groups and the A-E, in checking and analyzing data, in drafting and editing the report sections, and in developing the computer program used in carrying out the system analysis. In all, preparation of the report has taken approximately two years.

This report includes the quantities and characteristics of wastes produced in the post-fission fuel cycle. Both facility operations and facility decommissioning wastes are described.

Facilities for treating and managing spent fuel, high-level and transuranium (TRU) wastes, and post-fission airborne and gaseous wastes are then described in sufficient detail to enable environmental impact assessments to be made for three nuclear fuel cycles: 1) the once-through cycle, 2) the uranium-only recycle, and 3) the uranium-plutonium recycle; and for two variations: 1) the deferred cycle, where the decision for disposal or reprocessing of spent fuel is delayed for a number of years and 2) delayed repository availability. Waste management costs for these fuel cycles and safeguard requirements to prevent theft and diversion or misuse of spent fuel or radioactive wastes are also examined. The treatment and management of non-TRU wastes destined for disposal in shallow-land burial sites is not included in this report.

A brief generic description of the primary post-fission fuel cycle facilities, including a nuclear power plant, a fuels reprocessing plant, and a mixed-oxide (UO_2-PuO_2) fuel fabrication plant, is provided. It is not the purpose of this report, however, to examine the entire post-fission fuel cycle in detail; only the waste management aspects are emphasized. Thus, there is no attempt here to develop a comprehensive comparison of the merits of uranium recycle or uranium and plutonium recycle versus no recycle (once-through cycle).

The status of waste management technologies was assessed in ERDA-76-43, where it was concluded that " . . . all technologies needed to manage radioactive wastes from the backend of the commercial LWR fuel cycle are commercialized, available, or under development; there are no gaps." In fact, in almost all cases several technology alternatives could be used for each step in the waste management system.

Both individual waste management facilities and the entire waste management systems are analyzed. Steps in the waste management system are classified into four main categories: treatment, storage, transportation, and final isolation. The requirements for waste management in each category are examined in detail.

One or more of the available or commercialized technologies for each waste management step for each TRU waste type were selected for detailed analysis here. Commercialized technology is defined as technology in routine industrial use, while available technology is defined as technology that has been developed to the extent that design and construction of a full-scale commercial installation could be initiated, although design verification tests might be required. In the case of geologic repositories for waste isolation, the technology of mine construction is well developed but the technology of site characterization and site selection and the exact details of waste emplacement require further development.

Neither the technologies selected for the analysis nor the waste management facilities that are described are necessarily optimum ones; nor have all available technologies necessarily been analyzed. Those selected, however, are fully representative of technologies which could be used. Thus, analysis of environmental impacts can be used to draw generic conclusions concerning waste management in the post-fission commercial LWR fuel cycle. Detailed analysis of two or three alternatives of most waste management steps gives further confidence that the range of potential costs and environmental impacts is adequately evaluated in the generic impact statement. In actual applications it is reasonable to expect that there could be some

improvement over the concepts described here that might be reflected in either more efficient waste management or lower environmental impacts or both.

It is recognized that in some cases the technologies being compared for a given task are not equally well developed; thus, more extensive data are available in some instances than in others. In all cases, however, sufficient information is available to permit reasonable environmental and cost evaluations.

This report is summarized in Section 1 and the bases and assumptions for the analyses developed here are discussed in Section 3. Waste treatment alternatives for the TRU wastes produced in the recycle alternatives are discussed in Section 4. Interim storage of packaged TRU wastes and spent fuel are described in Section 5, as are the facilities required for packaging (encapsulating) spent fuel in the event it is declared to be a waste. Transportation alternatives are described in Section 6, and the final isolation alternatives are described in Section 7. Section 8 contains a discussion of decommissioning alternatives and definitions of the quantities of decommissioning wastes. A brief discussion of thorium fuel cycle wastes is presented in Section 9.

An analysis, including costs, of complete waste management systems for all fuel cycles is presented in Section 10. The systems analysis in this section includes the waste management requirements for a projected installed nuclear capacity of 400 GWe in the year 2000 and for the operation of this system throughout the 40-year useful life of the projected nuclear power plants. Disposition of the TRU wastes produced as a result of decommissioning the nuclear fuel cycle facilities for this system is also analyzed.

The startup dates for nuclear fuel cycle facilities in the system analysis scenarios were selected early in the study and some of the early dates are no longer realistic possibilities. However, alternative cases serve to bracket a broad range of possible startup dates and show that these dates are not critical to either waste management costs or environmental effects. For example, cases are developed with geologic repository startup dates of 1985 and 2000 and reprocessing startup dates of 1981 and 2010.

Acronyms are used extensively throughout this report to avoid repetition of lengthy identifications. A list of acronyms used and their definitions is provided at the end of each volume.

The metric system of measurements as defined by the International System of Units (SI) is used as the primary measurement system. In the text, the quantities in SI units are generally followed by the quantities in common english units in parentheses, while measurements in tables and figures are generally given only in SI units. A table of conversion factors is provided at the end of each volume.

The term "reference" has several different applications in this report and a definition of these applications is provided here to aid readers in avoiding misinterpretation:

1. When two or more technology alternatives are available for accomplishing a waste management function, for example, vitrification or calcination for high-level waste solidification or indoor and outdoor methods for low-level waste storage, one alternative is selected as the "reference alternative."

2. The specific process or concept chosen to implement the "reference alternative" is the "reference process" or "reference concept"; for example, spray calcination/in-can melting is the "reference process" for the vitrification "reference alternative." The selection of a "reference process" or "reference concept" is done so that emissions, secondary waste quantities, and waste management system costs can be defined.

3. The waste management system composed of the "reference alternatives" is referred to as the "reference system."

4. The spent fuel composition used as a basis for defining the radionuclide content of the various wastes is referred to as the "reference fuel."

3.0 BASES AND BACKGROUND INFORMATION

3.0 BASES AND BACKGROUND INFORMATION

This section describes the assumptions, estimates, and calculations that were required to provide a basis for describing the waste management facilities discussed in subsequent sections. The following topics are covered:

Section 3.1, Fuel Cycle Options, describes the scope of fuel cycle alternatives considered.

Section 3.2, Primary Fuel Cycle Facility Descriptions, describes the reference generic facilities where the primary wastes originate. These include the nuclear power plant, the independent spent fuel storage basin, the fuel reprocessing plant and the mixed oxide fuel fabrication plant.

Section 3.3, Waste Descriptions and Classifications, describes and classifies the estimated quantities and characteristics of the primary radioactive wastes that originate in the primary facilities. Detailed radioisotope composition and decay data that are based on reference fuel burnup calculations are provided.

Section 3.4, Waste Management Alternatives, describes with a series of flow charts, the waste management alternatives and sequence of operations considered. The reference alternatives selected for the system calculations are also identified.

Section 3.5, Secondary Wastes, describes the estimated quantities and characteristics of the secondary TRU wastes produced in the reference waste management system.

Section 3.6, Basis for Facility Descriptions, describes how the waste management facility descriptions were developed and describes the format and content of these descriptions.

Section 3.7, Accident Analysis Basis, describes the scope of accident analysis, the method of classifying accidents for the subsequent consequence analysis in DOE/ET-0029, and the format for describing postulated accidents.

Section 3.8, Cost Analysis Basis, describes the assumptions, scope, and methodology used in the development of the waste management facility capital, operating and levelized unit costs.

Section 3.9, Basis for Safeguards and Physical Protection Requirements, describes the methodology used in analyzing the safeguards and physical protection requirements for each of the waste management facilities.

Section 3.10, Fuel Cycle Projections, describes the nuclear power growth projection, spent fuel mass flows, and the number and schedule of primary fuel cycle facilities required for each of the fuel cycle cases described in Section 3.1.

3.1 FUEL CYCLE OPTIONS

3.1 FUEL CYCLE OPTIONS

The selected fuel cycle option determines the types and quantities of radioactive waste produced and the technological alternatives and facilities required for waste management. This section describes the fuel cycle options for which these waste management processes and facilities are described in this report. The waste management facilities and alternatives for implementing these fuel cycle options are described in Sections 4-8. Only light water reactor (LWR) uranium fuel cycles are considered in detail. However, in Section 9 a qualitative discussion of thorium fuel cycles and their wastes is presented. The waste management requirements for the entire system consisting of all of the LWR nuclear power plants projected for construction in the United States by the year 2000 are described in Section 10 for each of the fuel cycles described here.

A number of LWR fuel cycle variations are conceivable in addition to those presented. However, the cycles discussed here are representative of proposed LWR fuel cycles and provide a broad basis for evaluating potential environmental impacts of waste management.

3.1.1 Once-Through Cycle

The once-through fuel cycle assumes isolation of irradiated fuel elements as a waste without reprocessing for recovery of contained energy values. This concept is identified in this report as the Case 1 Fuel Cycle. In this cycle irradiated fuel will be stored until a qualified Federal waste isolation repository is available. Interim storage is provided either at the nuclear power plant or at an offsite independent spent fuel storage facility (ISFSF) if sufficient storage capacity is not available at the power plant. If an ISFSF is used, a short storage period at the power plant is required to simplify transportation and other subsequent handling steps. Small quantities of non-transuranic, airborne and gaseous wastes will be produced at the ISFSF. The largest quantities of waste other than spent fuel will be generated during decommissioning of the fuel cycle facilities.

To permit the spent fuel to cool sufficiently for dry packaging (encapsulation) for final isolation, this fuel cycle provides 6.5 years storage of unpackaged spent fuel in water-cooled storage basins. During this storage period it is assumed 75% of the unpackaged spent fuel is stored in onsite nuclear power plant water basins with the remainder stored at ISFSF's. Following the 6.5 years of interim storage, the spent fuel is packaged and sent to final isolation (assumed here to be a geologic repository). Facilities for packaging spent fuel are assumed to be located at an ISFSF site. Transportation of the spent fuel is required between the onsite and offsite storage-basin-packaging facilities and between the storage-basin-packaging facilities and a repository. Figure 3.1.1 is a simplified diagram of the once-through cycle.

For the Case 1 Fuel Cycle a geologic repository is assumed to be available for packaged spent fuel disposal in 1985.

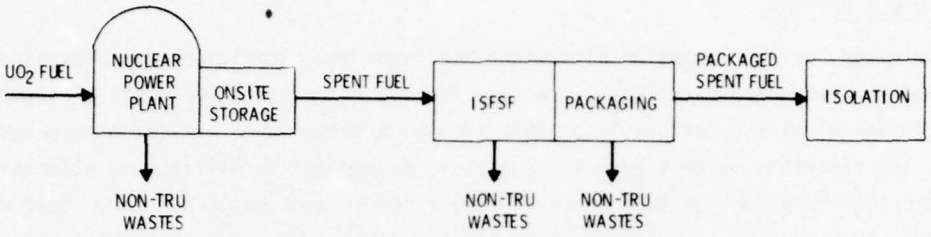

FIGURE 3.1.1. Once-Through Fuel Cycle (Case 1)

3.1.2 Uranium-Only Recycle

Uranium-only recycle assumes that spent fuel will be reprocessed to recover the residual uranium for recycle without plutonium recycle. This concept is identified in this report as the Case 2 Fuel Cycle. The recovered plutonium is either treated as waste (Case 2A) or stored for use in other reactor types at some future date (Case 2B).

As in the once-through cycle, the spent fuel in the uranium-only recycle may either be stored onsite at a nuclear power plant or offsite at an ISFSF. However, the storage requirements are much lower than in the once-through case, and it is assumed that all storage requirements can be met by nuclear power plant storage basins and reprocessing plant basins. Subsequently, the spent fuel is transferred to the reprocessing plant, where it is again stored briefly prior to reprocessing. During reprocessing, uranium and plutonium are separated. The uranium is returned to a uranium enrichment plant as UF_6 and is recycled to nuclear power plants in the form of enriched UO_2 fuel. Separated plutonium is either stored in an oxide form for future use or disposal or is combined with high-level liquid waste during solidification and ultimately sent to final isolation (assumed here to be a geologic repository). Various transuranic wastes are produced during the reprocessing step. In addition, non-transuranic and various airborne and gaseous wastes are generated at the storage basins and reprocessing plants. Figure 3.1.2 is a simplified diagram depicting the uranium-only recycle modes.

Prior to reprocessing, the reference cycle assumes storage periods of a minimum one-half year at the nuclear power plant and up to one year at the fuels reprocessing plant.

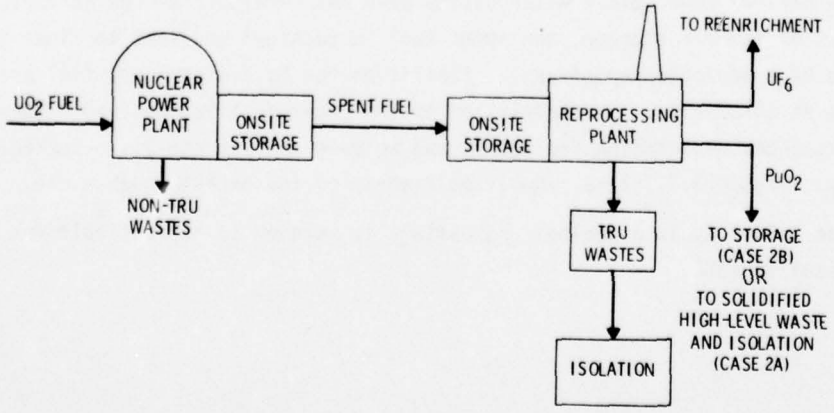

FIGURE 3.1.2. Uranium-Only Recycle (Case 2)

Reprocessing is assumed to start in 1981 and a geologic repository is available for the TRU wastes in 1985.

3.1.3 Uranium and Plutonium Recycle

Recycle of plutonium in addition to uranium allows recovery of the energy values associated with the plutonium. This concept is identified in this report as the Case 3 Fuel Cycle. Except for the reuse of plutonium, the uranium-plutonium recycle option corresponds to the uranium-only recycle option. Rather than storing plutonium as an oxide or incorporating it with solidified high-level wastes, it is transferred from the reprocessing plant to a mixed-oxide fuel fabrication plant (MOX FFP). The mixed-oxide fuel from the MOX FFP is then returned to a nuclear power plant for additional power generation. In addition to the TRU wastes resulting from the uranium-only recycle option, TRU wastes are generated at the mixed-oxide fuel fabrication plant. The transuranic concentration in wastes produced during recycle of uranium and plutonium will be greater than that in the uranium-only cycle because of the higher transuranic concentrations in recycle fuels. Figure 3.1.3 diagrams the uranium-plutonium recycle.

As in the uranium-only cycle, the uranium and plutonium recycle assumes storage periods of at least one-half year at the nuclear power plant and up to one year at the fuel reprocessing plant. Reprocessing and geologic repository availability dates are the same as for Case 2, i.e., 1981 and 1985 respectively.

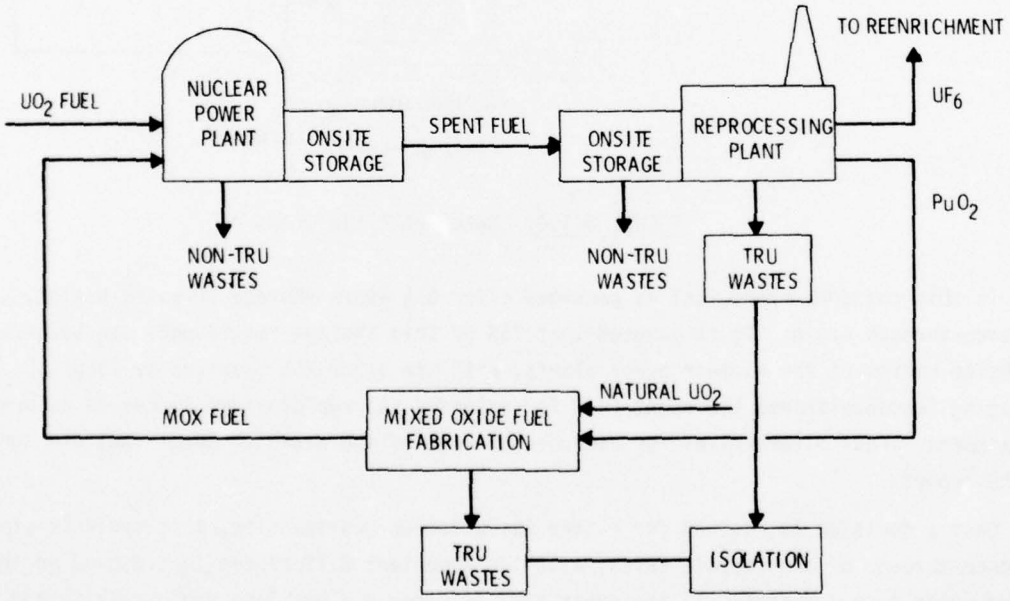

FIGURE 3.1.3. Uranium-Plutonium Recycle (Case 3)

3.1.4 Deferred Decision to Dispose or Reprocess Spent Fuel

A deferred decision to dispose or reprocess spent fuel elements, unlike the once-through cycle, provides a strategy that does not foreclose, through direct disposal, future options to recover the potential energy of the uranium and plutonium contained in the spent fuel elements. By utilizing extended storage of spent fuel time is provided to make a more informed decision on the advisability of reprocessing the spent fuel. This concept is identified in this report as the Case 4 Fuel Cycle. Once a decision is made, spent fuel can either be sent to final isolation (Case 4A), as in the once-through cycle, or sent to a reprocessing plant (Case 4B), as in the recycle concepts. Figure 3.1.4 is a simplified diagram of the deferred cycle.

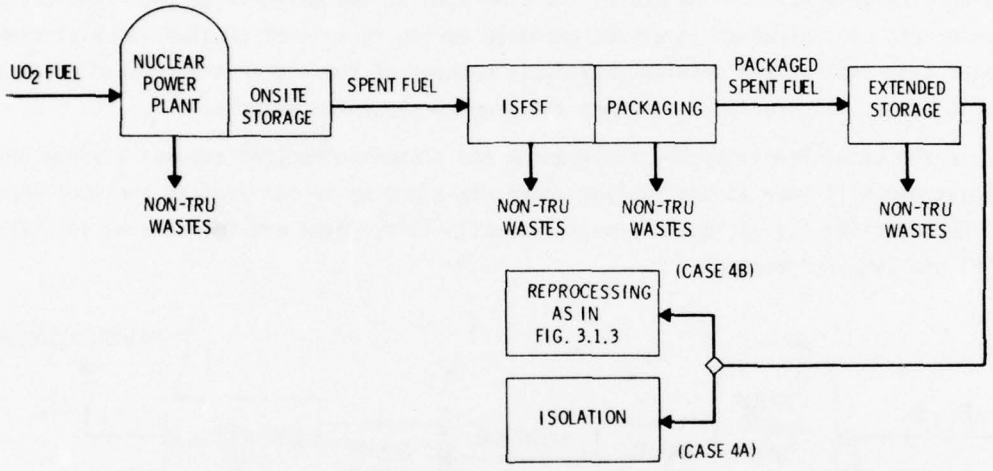

FIGURE 3.1.4. Deferred Cycle (Case 4)

In this concept, spent fuel is packaged after 6.5 years storage in water basins, as in the once-through cycle. It is assumed that 75% of this storage requirement can be provided by onsite basins at the nuclear power plants, with the other 25% provided by ISFSF's. Packaging (encapsulating) the spent fuel for extended storage provides increased assurance of containment. Four alternatives for extended storage of the packaged spent fuel are considered in the report.

Once a decision is reached for either isolation or reprocessing, this cycle is similar to the once-through or the recycle cases, with two important differences that depend on the time period prior to a decision: 1) the spent fuel is older and has less radioactivity and 2) the accumulation of spent fuel in storage facilities may be quite large and may take a number of years to work off. These effects are discussed in more detail in Section 10.

Wastes generated in the deferred case are expected to be similar to those generated in the once-through cycle or in one of the reprocessing cycles. However, the deferred cycle wastes may differ from the others because of radioactive decay during the lengthened storage period.

For the Case 4 Fuel Cycle it is assumed that a decision is made in year 2000. In Case 4A a repository is assumed to be available by that date and spent fuel shipments to the repository start immediately. In Case 4B a 10-year period is required before reprocessing can start at which time shipments to the repository also start.

3.1.5 Delayed Repository Availability

Another fuel cycle alternative considers the possibility of a substantial delay in the availability of a geologic repository. Repository availability in the year 2000 is assumed.

In the case of the once-through cycle the impact is an increased requirement for spent fuel storage. Although the reason for the delay is different, the effect on facility requirements is the same as a deferred decision to dispose of spent fuel as a waste (Case 4A) and a separate case is not developed.

In the case of reprocessing, the cycle proceeds as in the uranium and plutonium recycle case (Figure 3.1.3) except that substantial interim storage facilities are required to hold the packaged waste products until a geologic repository is available. This concept is identified in this report as the Case 3B Fuel Cycle.

3.2 PRIMARY FUEL CYCLE FACILITIES

3.2 PRIMARY FUEL CYCLE FACILITIES

Facilities in the postfission portion of the light water reactor (LWR) fuel cycle that are potential generators of radioactive wastes include nuclear power plants, independent spent fuel storage basins, fuel reprocessing plants and mixed-oxide (UO_2-PuO_2) fuel fabrication plants. To provide a basis for estimating quantities and characteristics of the radioactive wastes produced at these fuel cycle steps and to provide perspective for waste treatment costs, construction requirements, etc. relative to the primary facilities, it is necessary to define the basic features of plants that are generically representative of these primary fuel cycle facilities. The following descriptions of these generic primary facilities exclude those portions of the facilities required for radioactive waste management. Waste management equipment and operations within the primary facilities for treatment of transuranium (TRU) wastes and gaseous wastes (except those from the nuclear power plant) are described separately in subsequent sections. The initial pretreatment concentration operations for dilute aqueous wastes and the nuclear power plant gaseous wastes treatment facilities, however, are included as a part of the primary facilities.

3.2.1 Nuclear Power Plant

An LWR nuclear power plant generates electricity using steam produced by heat from the controlled nuclear fission of uranium and plutonium. The water used as a coolant and a moderator in such reactors has the normal distribution of hydrogen isotopes, and is thus termed "light water" to distinguish it from "heavy water," which contains a high concentration of the deuterium isotope and is sometimes used as a reactor coolant.

3.2.1.1 Selection of the Reference Nuclear Power Plant

The boiling water reactor (BWR) and the pressurized water reactor (PWR) are the types of LWRs projected to produce most of the nuclear energy in the United States up to the year 2000. In the BWR the primary coolant is converted to steam in the reactor core and is then expanded through a turbine generator unit to produce electricity. In the PWR the primary coolant is maintained as a pressurized liquid in the reactor core and transfers heat to a secondary water coolant. This secondary coolant is in turn expanded through a turbine generator unit to produce electricity.

A PWR capable of generating 1200 MWe has been selected as the reference nuclear power plant for this report. This reference plant is representative of larger sizes of LWRs commercially available. PWRs account for about two-thirds of the installed U.S. LWR electricity-generating capacity, with BWRs making up the remaining one-third. While PWRs and BWRs differ somewhat in design concept, the capital costs and construction requirements are similar.

3.2.1.2 Nuclear Power Plant Process

Figure 3.2.1 gives a simplified process flow diagram for the reference nuclear power plant. The reactor core contains 193 fuel assemblies, each containing 264 UO_2 rods. The core contains a total of 100,000 kg (220,000 lb) of slightly enriched UO_2 in the form of ceramic pellets within Zircaloy cladding. The plant coolant system circulates water to transfer the heat generated in

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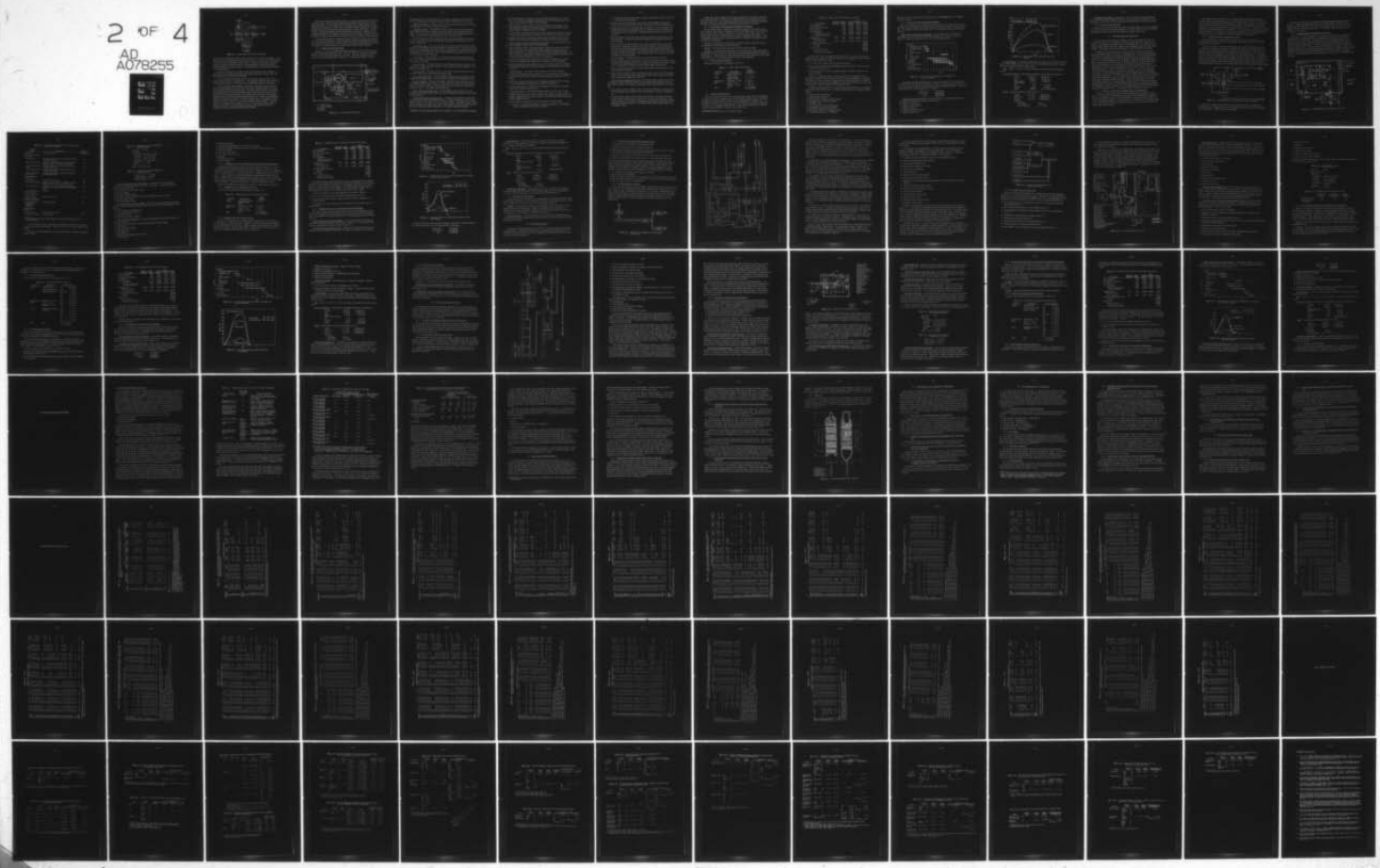

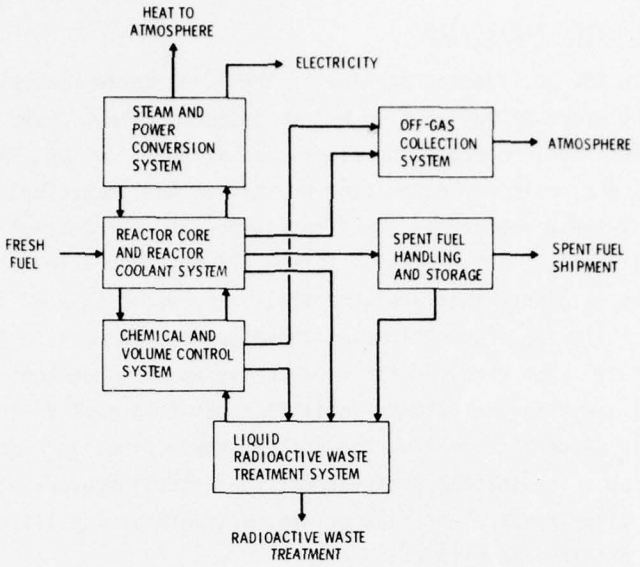

FIGURE 3.2.1. Nuclear Power Plant Process Flow Diagram

the core to the steam generators where steam is produced to drive the turbine generator. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber (boric acid) used for supplemental reactor control. Electricity is generated in the power conversion system by passing the steam from the steam generators through a high pressure turbine followed by low pressure turbines. The expanded steam is then condensed and returned to the steam generators.

A chemical and volume control system provides for maintaining the water inventory in the reactor coolant system and controlling water chemistry conditions, activity level, and soluble chemical neutron absorber concentration.

A spent fuel storage area receives the fuel from the reactor core and stores it for 6 months or more before it is shipped offsite. The operations conducted in this area are very similar to those discussed in Section 3.2.2 for the independent spent fuel storage basin.

A liquid radioactive waste treatment system receives liquid wastes from a variety of sources including chemical waste drain tanks, primary water storage overflow, fuel storage pool overflow, and assorted drains and sumps. The system includes filtration to remove particles and evaporation to yield a concentrated bottoms stream, containing the bulk of the chemicals and the radioactive materials, and a purified distillate stream. The condensed evaporator distillate may also be passed through demineralizers to remove entrained materials before being collected and analyzed. If analysis indicates that it is acceptable, the condensate can either be reused in the process or discharged to the environment. If the condensate is not acceptable for discharge or recycle, it is returned for further decay and additional processing. Some waste streams are analyzed after filtration only and, if acceptable, are then discharged without additional treatment.

An off-gas collection system comprises two parts, the gas collection system and the vent collection system. The gas collection system processes gaseous effluents that may contain appreciable amounts of fission product gases and hydrogen, but negligible amounts of oxygen; these gases are collected by a header and are then passed through a compressor and a moisture separator into waste decay tanks for holdup during decay of short-lived nuclides. The vent collection system purges gaseous effluents containing only low levels of radioactivity and hydrogen; these gases do not undergo any holdup period prior to release. The gases from both systems are filtered and then released into the auxiliary building ventilation system exhaust.

There are additional points of release of gaseous effluents that have at least the potential of containing contamination. These release points include the containment purge exhaust, the fuel and auxiliary building ventilation exhaust, the turbine building ventilation exhaust, the condenser air ejector exhaust, and the steam packing exhauster blower discharge.

3.2.1.3 Nuclear Power Plant Description

Figure 3.2.2 is a plot plan showing the general arrangement of the facilities that compose the reference LWR nuclear power plant. These facilities are located within a perimeter security fence enclosing approximately 8 ha (20 acres). The fence is centered within an exclusion area of approximately 400 ha (1000 acres). The main plant features include the

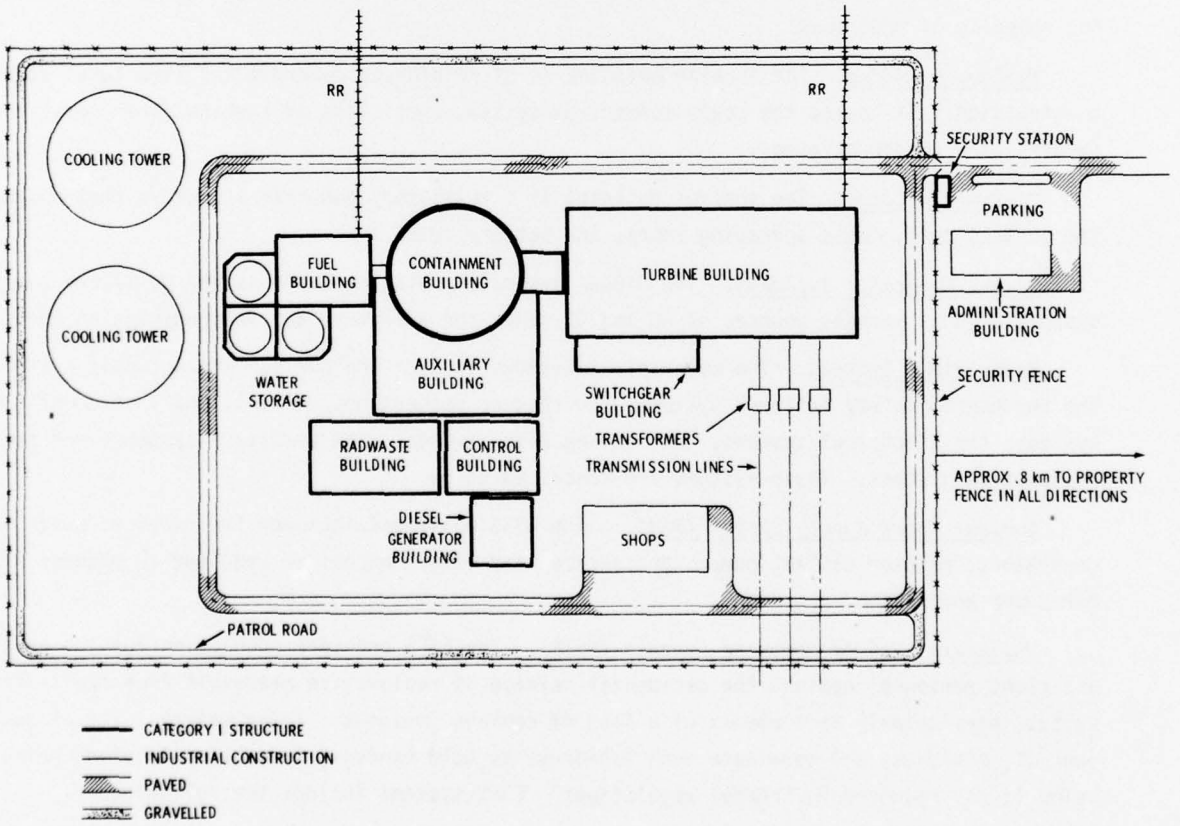

FIGURE 3.2.2. Nuclear Power Plant Plot Plan

containment, auxiliary, turbine control, fuel, switchgear, and diesel generator buildings. Heat is released through two natural draft cooling towers supplemented by a spray pond.

Major Plant Buildings. The major buildings and the functions performed within them are described below. All of these buildings are Category I.*

Containment Building. The containment building houses the reactor, steam generators, pumps, pressurizer, and other components that constitute the nuclear steam supply system (NSSS). This building is a post-tensioned concrete cylinder with a hemispherical dome. It is lined on the interior with welded carbon steel plate 6 cm (1/4 in.) thick.

The design objective of the containment building, including penetrations and isolation valves, is to maintain structural and leak-tight integrity during a design-basis loss of coolant incident (LOCI).

Auxiliary Building. The auxiliary building is a multistory concrete structure located adjacent to the containment building but physically separated from it. The auxiliary building encloses the vital components required for control and protection of the reactor during normal operation and in case of accident. These include engineered safety features, cooling water system components, and other equipment.

Fuel Building. The fuel building is a concrete structure that houses the spent fuel pool, a fuel receiving and storage area, and facilities for the cooling, handling, loading, and shipping of fuel casks.

Turbine Building. The turbine building is of reinforced concrete and structural steel construction. It houses the power conversion system, consisting of turbine generators, condensers, and condenser pumps.

Control Building. The control building is a multistory concrete structure that houses the control room, cable spreading rooms, and battery rooms.

Diesel Generator Building. The diesel generator building contains the auxiliary power systems used as standby sources of AC and DC power for emergency and instrumentation loads.

Major Plant Systems. The major plant systems include the nuclear steam supply system, the engineered safety features systems, the reactor protection, control, and instrumentation systems, the electrical systems, the cooling water systems, the auxiliary systems, and the radioactive systems. These systems are described below.

Nuclear Steam Supply System (NSSS). The NSSS equipment includes the reactor, steam generators, reactor coolant pumps, pressurizer and other components required to produce steam for power generation.

Engineered Safety Features Systems (ESFS). The ESFS provides protection for the public and plant personnel against the accidental release of radioactive materials from the reactor system, particularly as a result of a loss of coolant incident. These safety features localize, control, mitigate, and terminate such incidents to hold exposure to public and plant personnel below levels required by Federal regulations. ESFS systems include the following:

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* Designed to withstand maximum credible natural disasters, such as earthquakes and tornadoes.

- fuel building essential ventilation system, which detects radioactivity in the fuel pool if a fuel element is damaged during handling and initiates appropriate action to reduce release of fission products to the environment.
- containment building purge isolation system, which detects fission products inside the containment building following a fuel handling accident and initiates action to prevent release of radioactivity to the environment.
- control building essential ventilation system, which supplies the filtered air required to minimize the radiation dose to personnel in the control room following an incident.
- hydrogen purge system, which keeps the combustible gas concentration inside the containment to less than the lower combustible limit during a loss of coolant accident.

Reactor Protection, Control, and Instrumentation Systems. The reactor protection, control, and instrumentation systems consist of the following:

- reactor protection, which is provided through the inherent self-controlling characteristics of the reactor and through control element assembly positioning, control of boron concentration in the reactor coolant, and by the plant operating procedures. The reactor protection systems supply the operator with visual and audible alarms when any of the reactor parameters approach the limits of safe operation.
- reactor control systems, which provide for the startup and shutdown control of the reactor and for adjustment of reactor power in response to turbine load demands.
- nuclear instrumentation, which includes in-core and out-of-core neutron flux detectors and temperature sensors that provide information on flux and temperature distribution in the core.

Electrical Systems. The electrical system comprises the following subsystems:

- transmission and generating system, which consists of main generators, transformers and transmission lines that carry the power from the switchyard to the electric utility network.
- AC auxiliary system, which consists of two independent redundant sources of AC power for the ESFS equipment and standby power for normal plant operation.
- DC power system, which is provided by four independent units supplying the DC power required for instrumentation and control.

Cooling Water Systems. The following subsystems compose the cooling water systems:

- shutdown cooling system, which includes the cool down exchangers and pumps designed to reduce the temperature of the reactor coolant at a controlled rate during refueling and extended shutdown operations.
- cooling system for reactor auxiliaries, which include two separate systems:

  (1) essential cooling water system (ECWS), supplying cooling water to the components required to operate during and after an accident;

(2) nuclear cooling water system (NCWS), supplying cooling water to reactor auxiliaries during normal plant operations.

- essential spray pond system (ESPS), which consists of the equipment and components related to the spray pond and which provides cooling water to the ECWS described above.
- ultimate heat sink, which is provided by the spray pond and is used in conjunction with the ECWS and the ESPS during accident conditions. Makeup water for the ultimate sink, to replace water lost by evaporation, is the domestic water system and the plant cooling system.
- plant cooling water system (PCWS), which provides cooling water to the NCWS for normal plant operations. The cooling towers and the redundant heat exchangers and pumps are part of the PCWS.
- fuel pool cooling and cleanup systems, which provide forced cooling of the pool water during normal and emergency conditions (loss of offsite power). During normal operations the fuel pool heat exchangers are provided with cooling water from the NCWS, and during loss of power from the ECWS. The purification loop, which is part of the system, circulates cooling water through a filter and demineralizer unit to ensure cooling water that meets the design requirements.

Auxiliary Systems. The following subsystems constitute the auxiliary systems:

- chemical and volume control system (CVCS), by which the chemistry of the reactor coolant is maintained by continuous purification, filtration, and injection of additives. Volume control is maintained by adjusting the rate of coolant removal from the core.
- process sampling system, designed to collect samples remotely from the reactor coolant and auxiliaries for analysis.
- other auxiliary systems, including the heating, ventilation and air conditioning system (HVAC), the fire protection system, communication system, lighting system, and domestic water system that are common to all facilities.

Radioactive Waste Systems and Facilities. The radioactive waste systems and facilities are designed to control radioactive solid, liquid, and gaseous wastes produced at the LWR. Those systems included in the reference nuclear power plant concept developed for this report are:

- liquid radioactive waste treatment systems, which recover radioactive or chemical liquid waste and pretreat them by operations such as concentration, filtration, ion-exchange, etc.
- off-gas collection systems, which collect the high activity, short half-life off-gas from the reactor coolant, compress it and store it in tanks to allow for decay. After decay, the contents of the tanks are discharged through the plant ventilation system and released with low activity gases. All waste gases are filtered and monitored prior to environmental release.

Systems and facilities required for final waste treatment are not included in the reference nuclear power plant, in keeping with the intended scope of the primary facility descriptions (see Section 3.2). Also, since nuclear power plant wastes are considered nontransuranic (non-TRU), their associated treatment facilities are not discussed in this report.*

3.2.1.4 Nuclear Power Plant Operating and Maintenance Requirements

The nuclear power plant is assumed to be in operation about 80% of the time at close to 90% of capacity on the average (a 70% capacity factor) and to produce about 7.4 billion kWh of electric power per year. While in operation it runs 24 hours a day, 7 days a week. The remaining 20% of the time is devoted to activities such as maintenance and refueling. Scheduled shutdowns would tend to be as infrequent as possible. Operation of the spent fuel storage pool must be continuous.

Because of the high levels of radioactivity in certain portions of the plant, operation and maintenance of these portions must be done remotely.

Staffing. The reference plant would require a staff of about 100 people.

Utilities. The electric power consumption of the reference nuclear power plant would be about 40 MW. The rate of removal of liquid water from the environment would be about 1800 ℓ/sec (28,000 gal/min). About half of this water would be returned to the environment as water vapor, and essentially all of the remainder would be returned in the liquid form.

3.2.1.5 Nuclear Power Plant Emissions

Reference plant emissions are summarized in Table 3.2.1.

TABLE 3.2.1. Nuclear Power Plant Emissions

Emissions	Descriptions	Annual Quantity
Gaseous	Building ventilation exhausts (air)	1.5×10^{10} m^3
Liquid	Liquid effluents (water) to river	1.5×10^{10} kg
Cooling tower water:	Evaporated, T = 38°C	2×10^{10} kg
	Drift, T = 38°C	1×10^{8} kg
	Blowdown, T = 27°C	4×10^{9} kg
Other	Heat	1.5×10^{7} MW-hr (5×10^{13} BTU)

3.2.1.6 Nuclear Power Plant Costs

The total facility capital cost for the reference nuclear power plant was developed from an analysis of actual or estimated costs for a large number of domestic plants. Adjustments were made for plant power rating, mid-1976 licensing requirements, and mid-1976 costs. The breakdown of the total cost into man-hours, dollars, and quantities, as shown in Table 3.2.2, is typical of current generation BWRs and PWRs. No provision has been made for incremental costs resulting from changes in licensing criteria after 1976.

*Occasional TRU waste could arise from cladding failure.

TABLE 3.2.2. Nuclear Power Plant Capital Cost Estimate

Cost Element	Man-hours, 1000s		Costs, 1000s of Mid-1976 Dollars		
	Nonmanual	Manual	Material	Labor	Total
Major equipment		800	180,000	10,000	190,000
Buildings and structures		3,600	50,000	45,000	95,000
Bulk materials		3,200	50,000	40,000	90,000
Site improvements		400	10,000	5,000	15,000
Subtotal of direct site construction costs		8,000	290,000	100,000	390,000
Indirect site construction costs	3,000	2,000	35,000	60,000	95,000
Total field cost	3,000	10,000	325,000	160,000	485,000
Architect-engineer services					75,000
Subtotal					560,000
Owner's cost					240,000
Total facility cost					800,000
Estimated accuracy range					±5%

The total facility cost includes all plant-related costs incurred from the start of engineering to the initiation of commercial operation with the exception of working capital and the following specific exclusions:

- cost of the initial reactor core and subsequent refuelings
- costs of onsite radioactive waste management systems and facilities other than those specified in Section 3.2.1.3
- costs of shipping casks and other radioactive material containers intended for use primarily offsite.

Inasmuch as a substantial number of LWRs have been built, the engineering scope required for a functional plant consistent with recent licensing requirements is relatively well understood. Hence, the accuracy range primarily reflects uncertainties in quantities and pricing for labor, materials, and equipment. Within the indicated accuracy range, there is approximately an equal likelihood of cost overrun or underrun.

The operating and maintenance cost of the reference nuclear power plant is estimated to be about $20 to $25 million per year. In addition, fuel would cost an estimated $35 million based on:

- natural uranium at $40/lb U_3O_8
- separation work at $100/kg
- U_3O_8 conversion to UF_6 at $3.5/kg of uranium
- fabrication cost at $100/kg of uranium
- 3.2% ^{235}U in fuel
- 0.2% ^{235}U in tails from enrichment process
- 33,000 MWd/MT equilibrium fuel exposure
- 32% thermal to electric conversion efficiency
- a 70% capacity factor (7.4×10^9 KWh/yr).

These costs do not include any spent fuel storage costs, waste management costs, or reprocessing costs or credits.

3.2.1.7 Nuclear Power Plant Construction Requirements

Many factors relating to site preparation and construction of the reference plant may have some impact on the environment, the local economy, and the natural resources of the surrounding area. The information that follows provides a basis for evaluating the impact of construction activities.

Project Schedules and Construction Manpower. The estimated schedule for engineering, procurement and construction of the reference nuclear power plant is given in Figure 3.2.3. Figure 3.2.4 shows construction labor force size, composition and schedule.

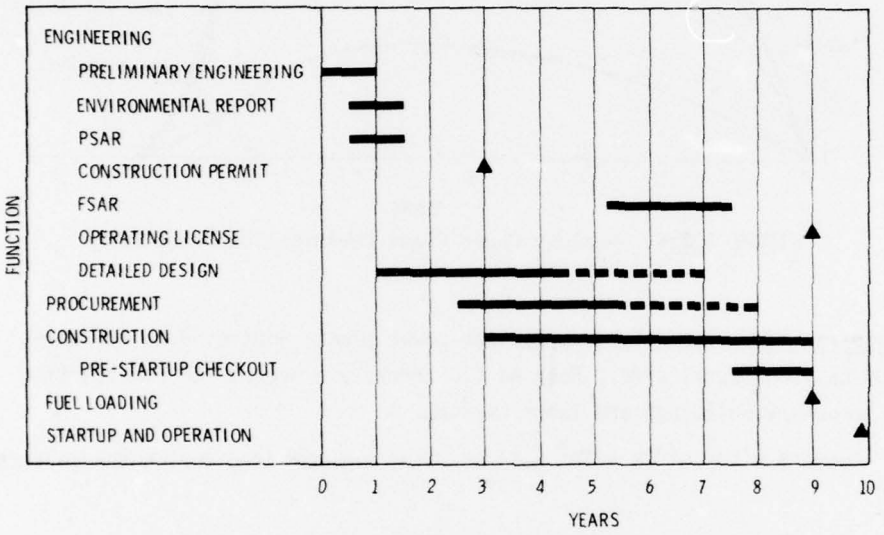

FIGURE 3.2.3. Nuclear Power Plant Engineering, Procurement and Construction Schedule

Distribution of Onsite and Offsite Costs. Onsite costs are those for all construction, materials, and services provided at the site of the plant while offsite costs are those for all services provided, and materials and equipment fabricated or assembled elsewhere. The distribution of total costs in these categories is as follows:

Onsite costs	$172,000,000
Offsite costs	$628,000,000
Total	$800,000,000

Temporary Construction Facilities. Temporary facilities and services required at the site during the construction period include the following:

- equipment storage areas
- temporary roads and fences
- temporary buildings
- electrical and water utilities
- temporary sewage service.

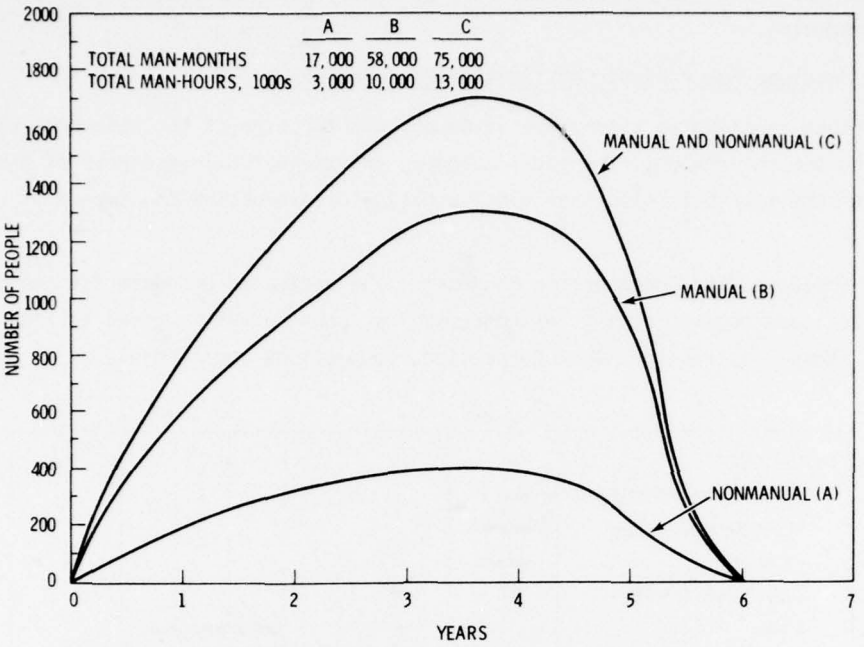

FIGURE 3.2.4. Nuclear Power Plant Construction Labor Force Schedule

Site Requirements. The reference nuclear power plant requires 8 ha (20 acres) for facilities on a 400 ha (1000 acre) site. Four ha (10 acres) are needed for construction storage, work yards, temporary buildings and labor parking.

Water. About 2×10^5 m^3 (5×10^7 gal) of water are required during the construction period.

Construction Materials. Major material requirements for facility construction are:

Concrete	190,000 m^3	(250,000 yd^3)
Steel	54,000 MT	(60,000 tons)
Copper (mainly wire and cable)	1,800 MT	(2,000 tons)
Zinc	9 MT	(10 tons)
Aluminum	450 MT	(500 tons)
Lumber	5,900 m^3	(2.5×10^6 board feet)

Energy. Resources used during construction will be:

Propane	1.5×10^6 ℓ	(400,000 gal)
Diesel fuel	1.5×10^7 ℓ	(3,900,000 gal)
Gasoline	1.0×10^7 ℓ	(2,600,000 gal)
Electricity		
Peak demand	2700 kW	
Total consumption	1.0×10^7 kWh	

Transportation Requirements. Approximately 1.6 km (1 mi) of new, two-lane paved road will be required to provide automobile and truck access from the nearest highway to the site. Approximately 3.2 km (2 mi) of new railroad spur will be required for site railroad service.

3.2.1.8 Effects of Fuel Cycle Options

Operation of a given nuclear power plant would be affected relatively little by implementation of any of the fuel cycle options considered in this study, namely, the no recycle option, the uranium-only recycle option, and the uranium and plutonium recycle option.

3.2.2 Independent Spent Fuel Storage Basin

Until such time as spent fuel can be shipped to either a fuel reprocessing plant or a Federal waste repository, a large amount of spent fuel storage capacity is expected to be required. Nuclear power plant storage basins can accommodate most of this spent fuel, but a sizeable portion (possibly 25%) may be stored in large independent spent fuel storage basins (ISFSBs). These storage basins are also referred to as Away From Reactor (AFR) basins.

A reference ISFSB is described here to provide a basis for defining the radioactive wastes and effluents produced in such fuel storage operations. Under the conventions used in this report, such wastes are considered to be primary wastes if the storage precedes fuel reprocessing but secondary wastes if the storage occurs as part of the once-through fuel cycle. This is because secondary wastes are defined as being those produced during management of primary wastes and the spent fuel is considered a waste in the once-through cycle. Thus, the ISFSB described here is considered to be used for storage of fuel before it is reprocessed. A ISFSB of slightly different design is described in Section 5.7.1 for use in the once-through cycle. The facility designed in Section 5.7.1 has a fuel shipping and receiving facility design which is more amenable to expansion, utilizes six storage pools each capable of storing 500 MTHM (instead of a single 3000 MTHM storage basin) and employs a more compact storage array. The more compact array is made possible by the use of stainless steel rather than aluminum storage racks and canisters or baskets (stainless steel absorbs neutrons more efficiently than does aluminum). Either of the two designs could be used for providing storage prior to reprocessing or packaging for final isolation; the alternate design described in Section 5.7.1 was developed to allow expansion of the receiving capacity to 2000 MTHM/yr to match the capacity of a fuel packaging facility and to provide for modular addition of a packaging facility. This modified ISFSB is described in Section 5.7.2.

3.2.2.1 Selection of the Reference Independent Spent Fuel Storage Basin

The reference ISFSB provides storage of the discharged LWR fuel elements in a water basin. It was chosen as the reference concept because it is the established and proven concept for the storage of LWR fuel. Other alternatives for long-term storage of spent fuel include storage of packaged fuel in air-cooled vaults, in dry caissons, or in surface storage casks. These concepts, however, all require several years of prior storage for radioactive decay before they are feasible.

The reference ISFSB has the capacity to store 3000 MTHM of irradiated PWR and BWR fuel. The total number of fuel assemblies that may be stored at design basis capacity is approximately 10,000, comprising 4000 PWR and 6000 BWR assemblies. Reference fuel, upon receipt, will have been cooled a minimum of 180 days and have an average burnup of 29,000 MWd/MTHM. The ISFSB is equipped to receive and ship spent fuel in casks by both rail and truck. The capacity for handling (receiving plus shipping) spent fuel is 1000 MTHM/yr. The facility can thus be filled or emptied within three years or it can provide a six-year lag storage period with a steady-state annual throughput rate equal to 500 MTHM/yr or one-sixth of the total storage capacity.

3.2.2.2 Independent Spent Fuel Storage Basin Process

The primary functions of the ISFSB are to receive, store, and ship spent LWR fuel. The processes required for these operations are described in more detail in Section 5.7.1 and are summarized only briefly here.

Figure 3.2.5 presents a schematic representation of the operations involved at an ISFSB. This figure also illustrates the sources of the wastes that result from the operation of the reference ISFSB. Spent LWR fuel assemblies are received in shipping casks. These casks are immersed in water and the assemblies are transferred to storage baskets or racks. The baskets or racks are then moved (underwater) to the storage position in the water basin. Fuel assemblies identified as containing leaking elements are placed in special containers which are vented to an off-gas treatment system. During all these operations the water serves as both a radiation shield and a heat transfer medium. After removal from the water basin, the empty casks are cleaned (decontaminated) both inside and out to specified levels and are then returned for another shipment.

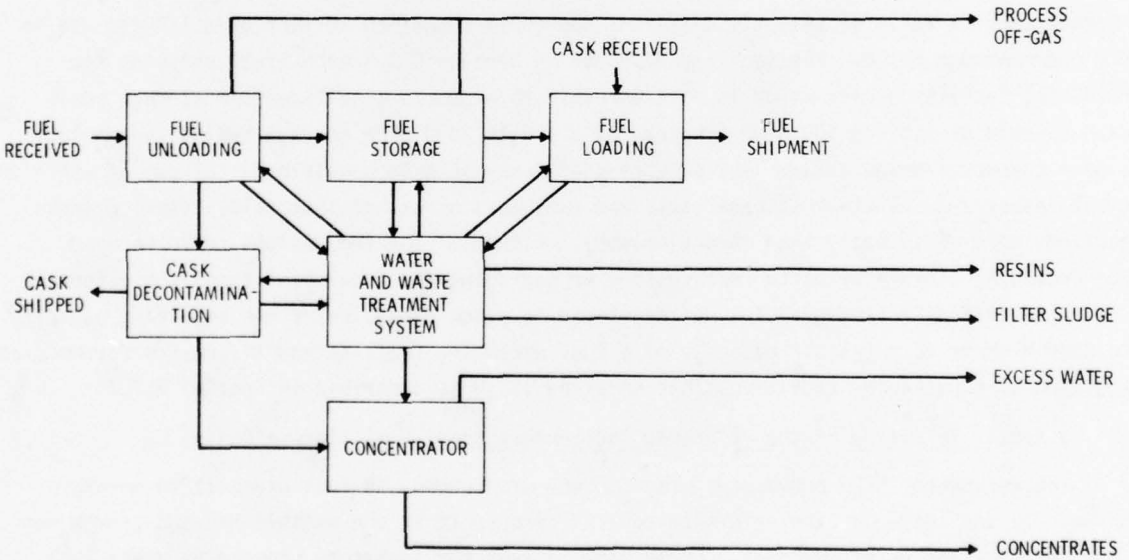

FIGURE 3.2.5. Independent Spent Fuel Storage Basin Process Flow Diagram

When fuel is shipped from the ISFSB to a fuel reprocessing plant, the assemblies are transferred from the storage baskets or racks back into shipping casks, which are then removed from the water basin, cleaned as required, and shipped.

The water within the basin is continuously recirculated through a treatment system to maintain its chemical and radiochemical purity within desired ranges. Solutions used to regenerate the ion exchange resins used in the water treatment system and some of the decontamination solutions are fed to a concentrator. The overheads from the concentrator are discharged to the atmosphere.

3.2.2.3 Independent Spent Fuel Storage Basin Description

Figure 3.2.6 is a plot plan showing the arrangement of the facilities that compose the reference ISFSB. The structural features of these facilities are listed in Table 3.2.3. The facilities are located within a perimeter security fence enclosing approximately 6 ha (15 acres) that, in turn, is centrally located within a 400 ha (1000 acre) site. Main plant features include the storage basin, cask and fuel handling area, and radioactive waste and basin water treatment area. These are enclosed within Category I structures. Support, utility, and general facilities are also provided within the secured area. Systems and features of the ISFSB associated with radioactive waste management are not considered for this reference facility. The approximate area that would be occupied by such facilities is indicated on the plot plan.

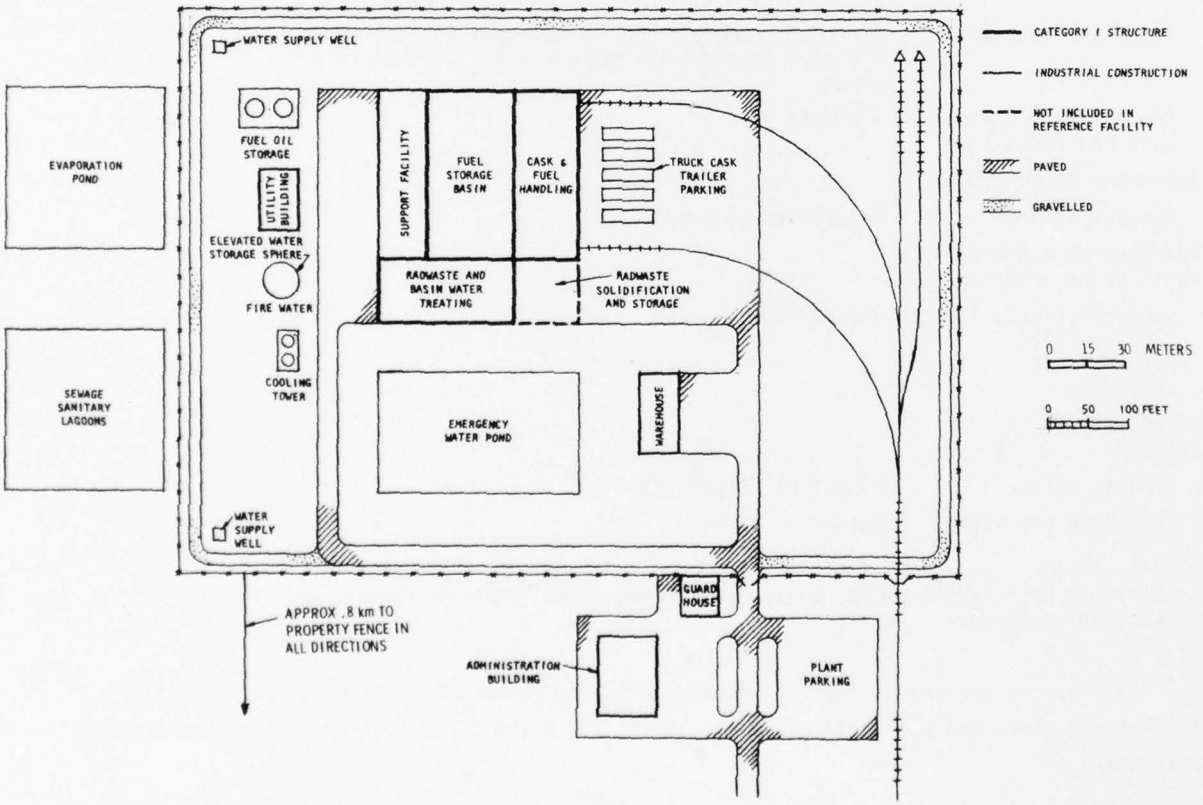

FIGURE 3.2.6. Independent Spent Fuel Storage Basin Plot Plan

TABLE 3.2.3. Description of Independent Spent Fuel Storage Basin and Related Structures

Structural Item	Description	Category I
Foundations	Reinforced concrete	As applicable
Cask and fuel handling facilities:		
Superstructure	Structural steel framing including crane girders and support columns	Yes
Roof deck and siding	Special insulated type with "blow-away" feature designed to fail in the event of a tornado	Yes
Cask unloading pools (CUPs)	Stainless steel lined, reinforced concrete constructed below grade	Yes
Fuel transfer canals	Stainless steel lined, reinforced concrete constructed below grade	Yes
Wash and decontamination pits	Stainless steel lined, reinforced concrete constructed below grade	No
Fuel storage basin facilities:		
Superstructure	Structural steel framing	Yes
Roof deck and siding	Special insulated type with "blow-away" feature designed to fail in the event of a tornado	Yes
Fuel storage basin	Stainless steel lined, reinforced concrete constructed to have the nominal water surface below grade	Yes
Fuel storage canisters and racks	Aluminum	Yes
Emergency power area:		
Superstructure	Reinforced concrete	Yes
Radioactive waste and basin water treating:		
Superstructure, including cells designed for shielding	Reinforced concrete	Yes
Support facility area:(a)		
Superstructure	Structural steel framing	No
Roof deck and siding	Insulated panels	No

a. Includes basin water pumps, cooling equipment, ventilation equipment and other support equipment.

Facility parameters of special importance to environmental considerations are those of the exhaust stack and the cooling tower. Tables 3.2.4 and 3.2.5 list typical values of such parameters.

Other major systems and equipment can be classified according to handling, storage, and support functions.

TABLE 3.2.4. Independent Spent Fuel Storage Basin Cooling Tower Data

Heat load	= 15 MW
Water flow:	
Circulating	= 310 ℓ/sec @ 27-38°C
Evaporated	= 6.2 ℓ/sec @ 38°C
In drift	= 0.03 ℓ/sec @ 38°C
In blowdown	= 1.1 ℓ/sec @ 27°C
Makeup	= 7.3 ℓ/sec

Note: 1 ℓ/sec = 15.8 gpm

TABLE 3.2.5. Independent Spent Fuel Storage Basin Exhaust Stack Data

Exhaust flow	= 120 m^3/sec
Exhaust velocity	= 15 m/sec
Release height	= 45 m

Cask and Fuel Handling Systems and Equipment. Irradiated fuel will be received in shipping casks, unloaded under water and placed in storage units. Cask and fuel handling systems and equipment include:

- cask, fuel and canister handling cranes
- wet cask cooldown
- cask wash and decontamination
- two cask unloading and loading pools (CUPs) approximately 15 m (50 ft) deep
- canister transfer canal
- cask and equipment storage.

Fuel Storage Basin Systems and Equipment. The fuel storage basin (FSB) has approximately 2000 m^2 (21,000 ft^2) of water surface and a depth of 9 m (30 ft). Major equipment and systems include:

- canister handling and storage crane
- fuel storage canisters and racks
- basin water cooling
- basin water cleanup and treatment.

Support Systems and Equipment. These systems and equipment provide the following functions for the ISFSB:

- heating, ventilating and air conditioning (based on once-through design)
- cooling water
- water supply and treatment
- steam generation
- compressed air
- commercial and emergency electrical power
- fire protection

- analytical laboratory
- administration and personnel service buildings and parking
- security (including gate house, yard fencing, lighting, intrusion alarms, etc.)
- roads and railroads
- cask truck trailer parking
- rail siding for cask rail cars
- warehouse
- sanitary waste disposal
- fuel oil storage.

3.2.2.4 Independent Spent Fuel Storage Basin Operating and Maintenance Requirements

Operating requirements may vary over a wide range from receiving spent fuel at rates up to 1000 MTHM/yr with no shipping operations, to passive storage of a full basin for long periods with neither receiving nor shipping activities, to shipping at rates up to 1000 MTHM/yr with no receiving operations. In any case, the facility is manned 24 hours a day, 7 days a week. To provide a basis for estimating representative waste generation compositions and rates, a reference operating condition has been defined assuming a full basin and receipt of spent fuel at an average rate of 500 MTHM/yr, with simultaneous shipments offsite at an average rate of 500 MTHM/yr. Under these conditions fuel is stored in the basin for 6 years.

Operating and maintenance requirements are discussed in further detail in Section 5.7.1.

3.2.2.5 Independent Spent Fuel Storage Basin Emissions

Emissions from the reference ISFSB are summarized in Table 3.2.6.

TABLE 3.2.6. Independent Spent Fuel Storage Basin Emissions

Emission	Description	Annual Quantity
Gaseous	Ventilation air and vaporized excess water	4×10^9 kg
Cooling tower water:	Evaporated, T = 38°C	2×10^8 kg
	Drift, T = 38°C	1×10^6 kg
	Blowdown, T = 27°C	3×10^7 kg
Other	Heat	1.3×10^5 MW-hr (4.5×10^{11} BTU)

3.2.2.6 Independent Spent Fuel Storage Basin Costs

The capital cost estimate for the reference ISFSB is shown in Table 3.2.7 expressed in mid-1976 dollars. The total capital cost includes all plant-related costs incurred from the start of engineering to the initiation of commercial operation with the exception of working capital and the specific exclusions stated below. A complete description of the cost estimate basis, assumptions, and definitions is given in Section 3.8.

TABLE 3.2.7. Independent Spent Fuel Storage Basin Capital Cost Estimate

Cost Element	Man-hours, 1000s		Costs, 1000s of Mid-1976 Dollars		
	Nonmanual	Manual	Material	Labor	Total
Major equipment		200	17,000	3,000	20,000
Buildings and structures		1,400	11,000	19,000	30,000
Bulk materials		800	8,000	10,000	18,000
Site improvements		100	1,000	1,000	2,000
Subtotal of direct site construction costs		2,500	37,000	33,000	70,000
Indirect site construction costs	700	500	12,000	16,000	28,000
Total field cost	700	3,000	49,000	49,000	98,000
Architect-engineer services					19,000
Subtotal					117,000
Owner's cost					33,000
Total facility cost					150,000
Estimate accuracy range					±30%

The estimate's accuracy range reflects uncertainties in the engineering scope required to provide a fully functional plant based on the technology described and in the pricing and quantities for labor, materials, and equipment. A contingency covering these and similar factors has been included in the base estimate. With the contingency included, there is an approximately equal likelihood of the indicated cost being overrun or underrun.

Two general categories of costs are excluded from this estimate. These are:

- costs of final radioactive waste immobilization and packaging facilities
- costs of shipping casks and other radioactive material containers intended for use primarily offsite.

The operating cost of the reference ISFSB is estimated to be in the range of $2 to $4 million per year.

3.2.2.7 Independent Spent Fuel Storage Basin Construction Requirements

Many factors relating to site preparation and construction of the ISFSB may have some impact on the environment, the local economy, and the natural resources of the surrounding area. The information that follows provide a basis for evaluating the impact of construction activities.

Project Schedules and Construction Manpower. The estimated schedule for engineering, procurement and construction of the ISFSB is shown in Figure 3.2.7. The construction labor force size, composition and schedule is shown in Figure 3.2.8.

Distribution of Onsite and Offsite Costs. Onsite costs are those for all construction, materials and services provided at the site of the ISFSB, while offsite costs are those for

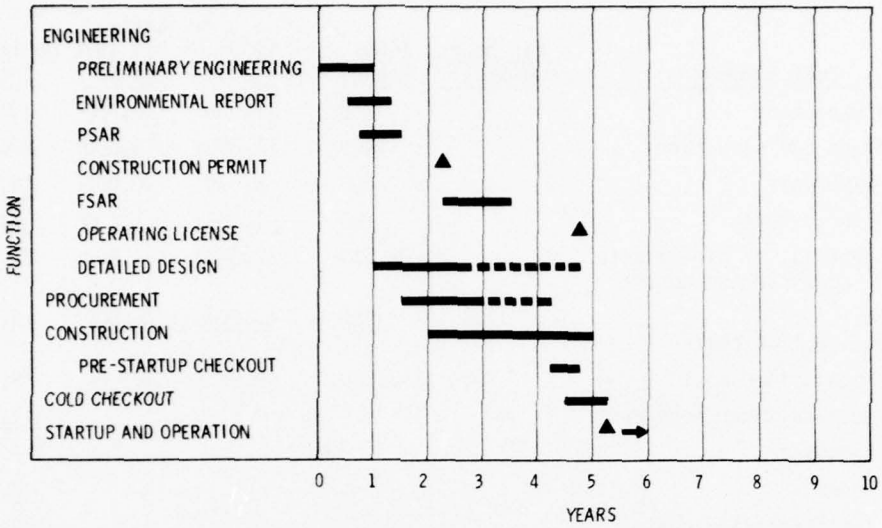

FIGURE 3.2.7. Independent Spent Fuel Storage Basin Engineering, Procurement and Construction Schedule

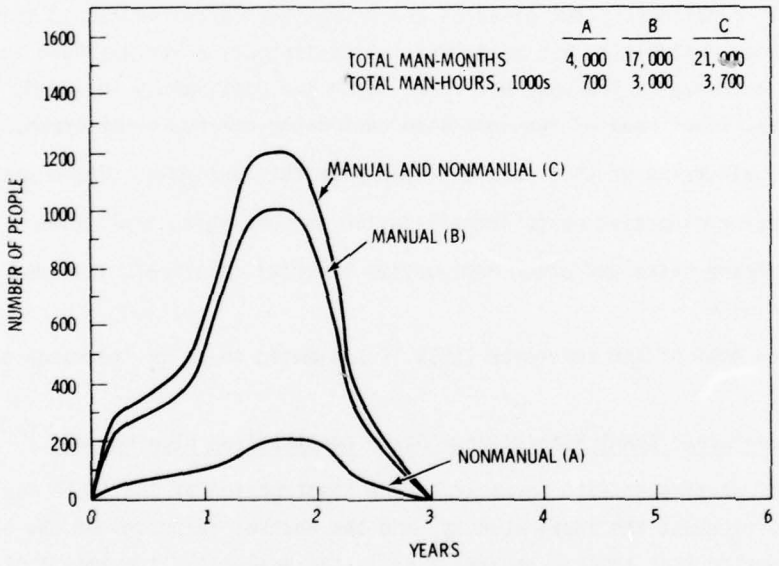

FIGURE 3.2.8. Independent Spent Fuel Storage Basin Construction Labor Force Schedule

all services provided, and for materials and equipment fabricated or assembled elsewhere. The distribution of total costs in these categories is shown below:

Onsite costs	$ 56,000,000
Offsite costs	$ 94,000,000
Total	$150,000,000

Site Requirements. The ISFSB requires 12 ha (30 acres) for facilities on a 400-ha (1000-acre) site. Approximately 4 ha (10 acres) are required for construction storage, work yards, temporary buildings, and labor parking.

Water. About 4.5×10^4 m^3 (1.2×10^7 gal) of water are required during the construction period.

Construction Materials. Major material requirements for facility construction are:

Concrete	23,000 m^3	(30,000 yd^3)
Steel	7,300 MT	(8,000 tons)
Copper (mainly wire and cable)	18 MT	(20 tons)
Zinc	9 MT	(10 tons)
Aluminum (racks and canisters)	820 MT	(900 tons)
Lumber	710 m^3	(300,000 board feet)

Energy. Resources used during construction will be:

Propane	380,000 ℓ	(100,000 gal)
Diesel fuel	4,200,000 ℓ	(1,000,000 gal)
Gasoline	2,600,000 ℓ	(700,000 gal)
Electricity		
Peak demand	700 kW	
Total consumption	3.8×10^6 kWh	

Transportation Requirements. Approximately 1.6 km (1 mi) of new road will be required to provide automobile and truck access from the nearest highway to the site. Approximately 3.2 km (2 mi) of new railroad spur will be required for site railroad service.

3.2.2.8 Effects of Fuel Cycle Options

The reference ISFSB was intended for use in supplementing nuclear power plant basin storage capacities when spent LWR fuel is to be reprocessed for recovery of uranium and plutonium. However it could be used in other fuel cycles with little change in operation.

No Recycle. Eliminating fuel recycle would do away with the use of an ISFSB for interim storage of fuel prior to reprocessing. However, the alternative would likely result in increased use of essentially identical storage basins for the interim storage of fuel prior to final disposition. This application is discussed in Section 5.7.1.

Uranium Recycle Only. This alternative would have little effect on the operation of the ISFSB other than reducing the decay heat release rate by about 10% because fewer heat-generating actinides would be present.

3.2.3 Fuel Reprocessing Plant

The purpose of the fuel reprocessing plant (FRP) is to recover and purify the uranium and plutonium present in spent fuels so that they can be recycled to a nuclear power plant. The reprocessing plant also provides facilities for converting uranium and plutonium nitrate solutions to UF_6 and PuO_2, respectively.

3.2.3.1 Selection of the Reference Fuel Reprocessing Plant

The reference FRP incorporates the following process steps:

- underwater storage of the spent fuel awaiting processing
- recovery and purification of the uranium and plutonium by solvent extraction (using the Purex process) to produce nitrate solutions of these metals
- conversion of the plutonium from nitrate solution to solid plutonium dioxide (PuO_2)
- conversion of the uranium from nitrate solution to solid uranium hexafluoride (UF_6).

Each of these steps has been carried out successfully for years in government facilities. No one plant has yet incorporated all of the steps; the FRP designs for Allied-General Nuclear Services (AGNS) and for Exxon Nuclear Company, however, include all these process steps, although with some differences in approach. Because of the general agreement on the best approach to be used (based on current technology), no other alternatives were considered.

The assumptions for operation of the reference FRP include:

- a processing rate of 2000 MTHM/yr
- spent fuel aged at least 6 months (after discharge from reactor) before receipt at the FRP
- spent fuel stored at the FRP for up to 1 year and at least 1.5 years old before processing
- spent fuel average burnup is 29,000 MWd/MTHM.

3.2.3.2 Fuel Reprocessing Plant Process

The reference FRP process more closely resembles the AGNS process than the Exxon Nuclear process. Figure 3.2.9 is a simplified block flow diagram showing the main process functions. The process steps are shown in more detail in Figure 3.2.10. This figure illustrates how "waste" streams from many operations are actually recycled to other operations for reuse, thus greatly reducing the overall amount of actual waste that must be treated. Figure 3.2.10 also illustrates the sources of the liquid and solid wastes that result from the operation of the reference FRP.

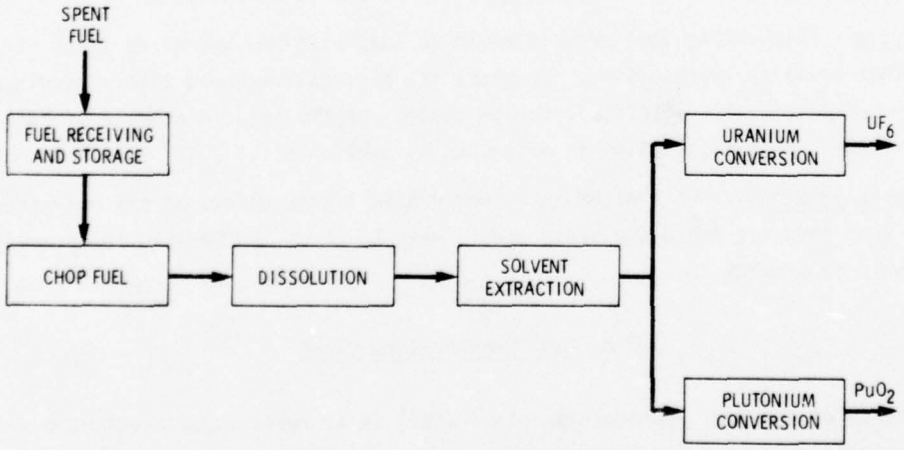

FIGURE 3.2.9. Simplified Flow Diagram of Fuel Reprocessing Plant Process

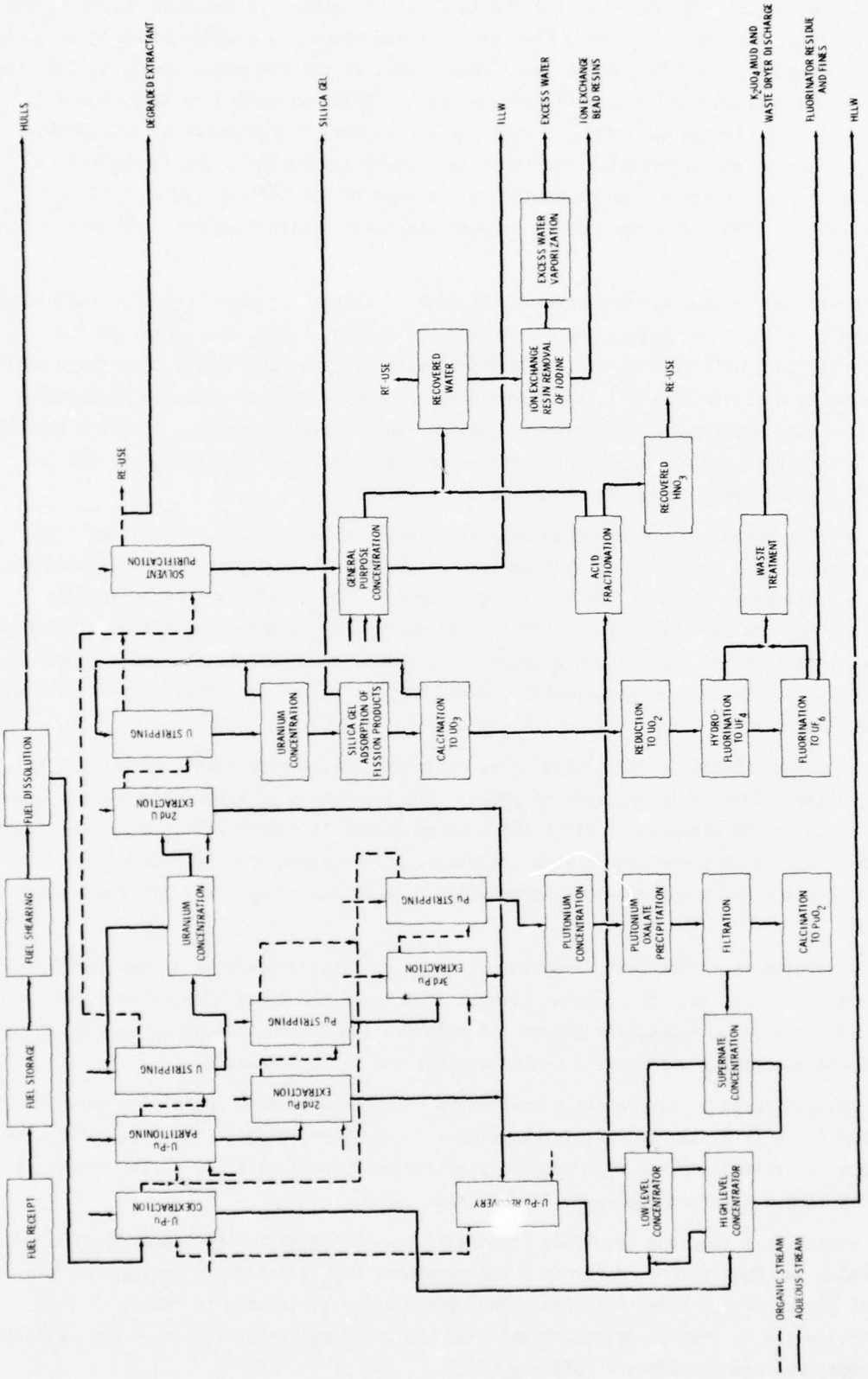

FIGURE 3.2.10. Flow Diagram of the Fuel Reprocessing Plant

Spent fuel assemblies arrive at the FRP in shielded casks. In the fuel receiving and storage facility, the fuel is removed from the cask and stored in a water basin in which the water is continuously purified and cooled. These steps at the FRP are essentially identical to those at the independent spent fuel storage basin (ISFSB) described in Section 3.2.2. Some receiving and storage operations at the FRP are different from those at the ISFSB, however, because of the opportunity for facility sharing at the FRP. The storage basin process off-gas, for example, can be routed to the same vessel off-gas system that serves the other FRP steps. Also, the same general purpose concentrator can handle liquid wastes from both sources.

The next step in the reference process is fuel shearing. In this step, the fuel elements are chopped into short segments a few centimeters in length so that the contained fuel (predominantly UO_2) will be exposed to the dissolvent. In the fuel dissolution step, nitric acid is used to dissolve the fuel, thus providing an aqueous nitrate solution containing nearly all of the uranium and plutonium and most of the fission products. The fuel cladding material (stainless steel or a zirconium alloy) does not dissolve in nitric acid and is removed as a solid waste (generally referred to as "hulls").

The uranium and plutonium are then separated from each other and are purified from fission products by a series of solvent extraction operations employing tributyl phosphate (TBP) in a hydrocarbon diluent. In the first contactor, the uranium and plutonium are coextracted into the solvent phase, which is then contacted (scrubbed) with a HNO_3 solution to improve separation of the fission products. Essentially all of the fission products leave this contactor in the aqueous phase, which, after concentration, is termed high-level liquid waste (HLLW).

In the second contactor the plutonium is preferentially stripped into an aqueous solution while the uranium remains in the solvent phase. The uranium is stripped into another aqueous solution in the third contactor. After the uranium stream is concentrated, a second solvent extraction cycle is performed to provide additional purification from plutonium and fission products. Final uranium purification is provided by sorption of residual fission products on silica gel.

The plutonium is further purified from uranium and fission products in two additional solvent extraction cycles. The aqueous streams from these two extraction columns and also from the second uranium extraction column are combined and passed through a recovery column to reduce the amounts of uranium and plutonium that end up in the waste.

The solvent phase is continuously treated to remove undesirable impurities that are formed, and the bulk of the solvent is recycled. The aqueous solutions used to purify the solvent are concentrated along with other miscellaneous aqueous streams. This concentrated solution is termed intermediate-level liquid waste (ILLW).

The nitric acid solution resulting from waste concentration is fractionated into "water" ($\sim$0.01$\underline{M}$ HNO_3) and HNO_3 ($\sim$10$\underline{M}$) fractions. The recovered HNO_3 fraction is reused, as is a portion of the recovered water fraction. The excess water is treated to remove fission product iodine and is then vaporized, mixed with the building ventilation air, and released to the atmosphere via the process building stack.

In the PuO_2 conversion facility, the plutonium is precipitated by the addition of oxalic acid and the plutonium oxalate thus formed is filtered off and is then calcined to plutonium dioxide (PuO_2).

In the UF_6 plant, the purified uranium is calcined to uranium trioxide (UO_3), which is reduced by hydrogen (cracked ammonia) to uranium dioxide (UO_2). The uranium dioxide is converted to uranium tetrafluoride (UF_4) by hydrogen fluoride, and the UF_4 is converted to uranium hexafluoride (UF_6) by reaction with fluorine in a fluidized bed.

To summarize, the following steps are involved in the reference FRP process:

- receiving and unloading spent fuel assemblies
- washing and decontaminating casks
- moving the fuel assemblies to the shear
- shearing the fuel assemblies
- dissolving the fuel
- separating the fuel hulls and hardware
- centrifuging the dissolver solution
- separating and purifying the uranium and plutonium in liquid-liquid solvent extraction contactors
- concentrating product and waste streams
- recovering nitric acid and water for reuse
- purifying the used solvent
- converting uranyl nitrate to UF_6
- packaging and storing UF_6
- converting plutonium nitrate to PuO_2
- packaging and storing PuO_2.

A schematic representation of the sources of the FRP gaseous waste streams is shown in Figure 3.2.11. The off-gases from the shearing and dissolution operations are combined for extensive treatment before being mixed with the building ventilation air. The vessels involved in the recovery and purification operations and most of the product conversion operations, as well as the gases released from leaking fuel elements in the fuel storage area, are vented to another system for treatment before being mixed with the building ventilation air. The ventilation air from the fuel receiving and fuel storage areas is released up a separate stack, while that from the remainder of the operations passes up the main stack.

Though the ventilation air from the fuel receiving and storage facility is normally released without treatment, an atmospheric protection system (APS) is installed for use in the unlikely event of serious contamination of this ventilation air. The air from nearly all of the other facilities is passed through an APS for final filtration before it is released up the stack. The exceptions to this are the air containing fluoride contamination from the operations involved in conversion to UF_6, and the vaporized excess water.

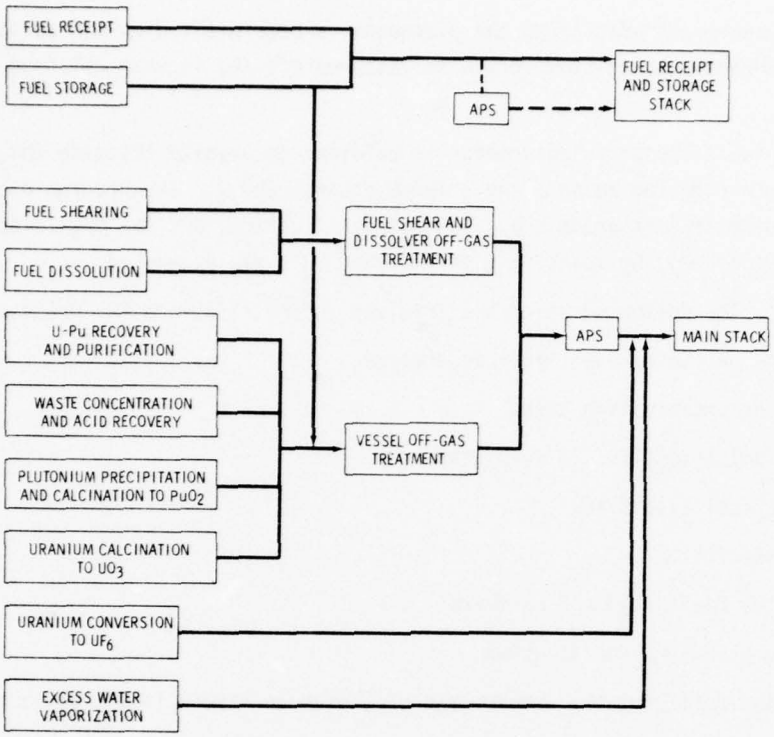

FIGURE 3.2.11. Sources of Fuel Reprocessing Plant Gaseous Waste Streams

3.2.3.3 Fuel Reprocessing Plant Description

The description of the FRP presented here is a general one, giving basic background information. It is meant to serve only as a point of reference for the more comprehensive sections that follow on radioactive waste management facilities. Waste management facilities sharing the FRP site, but not directly involved in the fuel reprocessing operations are:

- HLLW storage (Section 5.1), solidification (Section 4.1), and solidified high-level waste storage (Section 5.4)
- fuel residue packaging (Section 4.2) and storage (Section 5.2)
- failed equipment and noncombustible waste treatment (Section 4.3) and storage (Section 5.3)
- compactable and combustible waste treatment (Section 4.4)
- degraded solvent disposal (Section 4.5)
- ILLW immobilization (Section 4.7) and storage (Section 5.3)
- off-gas treatment and filtration (Sections 4.8-4.11)
- waste shipment, involving containers, casks, and transporters (Section 6).

The 2400-ha (6000-acre) plant site is fenced with posted agricultural-type fencing. The main plant facilities are located near the center of the site, within a protected security area of about 36 ha (90 acres).

Figure 3.2.12 is a plot plan for the reference FRP. The principal facility shown on the plot is the main process building, which encloses the head-end, solvent extraction and product conversion processes and is contiguous with the fuel receiving and storage building. These are seismic design Category I structures. Other Category I structures (indicated by heavy outlines on the plot plan) include the emergency utilities building, and the emergency water pond pump structure. Systems and features of the FRP associated with radioactive waste management are not included in the basic reference facility. They are described in subsequent sections as identified above. The approximate area that would be occupied by such facilities is indicated on the plot plan by dashed lines.

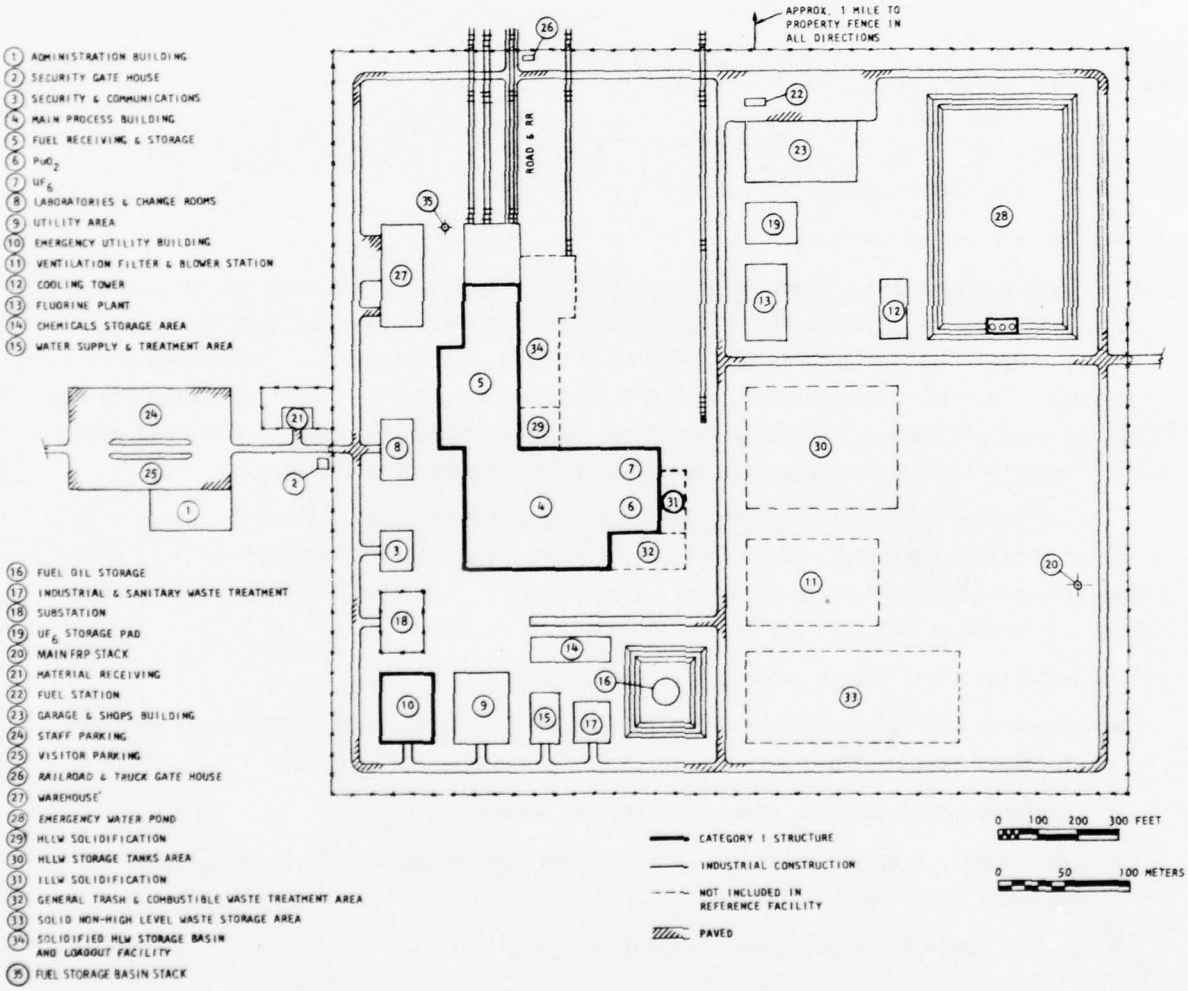

FIGURE 3.2.12. Fuel Reprocessing Plant Plot Plan

Main Process Building. The main process building houses most of the facilities requiring heavy shielding for protection of personnel from penetrating radiation. The cells and many other compartments of the process building are constructed of massive reinforced concrete walls up to 150 cm (5 ft) thick. For most cells containing radioactive materials, the floors and the lower portion of the walls are lined with stainless steel. Many of the cell and compartment surfaces not lined with stainless steel are covered with special radiation-resistant coatings and paint.

The main process building also includes:

- heating and ventilation systems
- sampling systems
- analytical laboratories
- health physics facilities
- control rooms
- maintenance cells
- remote operating and handling gear
- decontamination cells
- change rooms and offices.

Fuel Receiving and Storage Building. The fuel receiving and storage building has a structural steel frame, including crane supports and girders. The building design incorporates a special insulated "blow-away" siding and roof deck intended to fail in event of a tornado. The cask unloading pool, wash and decontamination pits, fuel transfer canals and fuel storage pool are below-ground structures made of reinforced concrete and lined with stainless steel. Fuel storage canisters and racks are made of aluminum.

Fuel Reprocessing Plant Utility and Other Support Facilities. The utility and other support facility buildings, other than the Category I structures mentioned earlier, are conventionally designed, with structural steel framing, insulated siding and roofs, and concrete foundations. Such facilities include:

- utilities plant (steam, water, air, electrical)
- emergency utilities, including two 100% capacity diesel electric generators and an emergency steam boiler
- substations and electric power distribution systems
- water supply systems, water treatment systems for process, utility and sanitary services, and water distribution systems
- cooling tower, with pumps and circulation system
- emergency cooling pond with pumps
- gaseous effluent discharge lines and two stacks (storage basin and main plant)
- fire protection facilities, including fire water pumps

- security facilities
- warehouse
- garage and shops building
- administration building
- sewers and sewage treatment system
- roads, rail spur tracks, and fences.

Cooling tower and exhaust stack data of particular importance to environmental considerations are given in Tables 3.2.8 and 3.2.9.

TABLE 3.2.8. Fuel Reprocessing Plant Cooling Tower Data

Heat load	= 75 MW
Number of cells	= 3
Water flow:	
Circulating	= 1500 ℓ/sec @ 27-38°C
Evaporated	= 31 ℓ/sec @ 38°C
In drift	= 0.15 ℓ/sec @ 38°C
In blowdown	= 5.4 ℓ/sec @ 27°C
Makeup	= 36 ℓ/sec
Note: 1 ℓ/sec	= 15.8 gpm

TABLE 3.2.9. Fuel Reprocessing Plant Exhaust Stack Data

	Exhaust Flow, m^3/sec	Exhaust Velocity, m/sec	Release Height, m
Main process stack	120	25	110
Fuel receipt and storage stack	80	15	45

3.2.3.4 Fuel Reprocessing Plant Operating and Maintenance Requirements

Operation of the FRP storage pool must be continuous, while operation of the other functions can be intermittent. Because of start-up and shutdown inefficiencies, however, it is planned that processing be continuous for extended periods of time, and that shutdowns be scheduled and infrequent. Maintenance at the FRP employs, for the most part, remote handling techniques. Cranes, manipulators, transfer carts, swing arms, baskets, racks, hatches, and viewing windows are routinely used as part of remote maintenance activities.

Staffing. The reference FRP would require a staff of about 1000 people excluding waste management functions.

Utilities. Fossil fuel to generate about 75 MW of heat would be consumed by the FRP utilities plant. The electrical power consumption of the reference FRP would be about 17 MW. The steam consumption of the FRP would be about 70,000 kg/hr (150,000 lb/hr). The steam would be obtained from the onsite utilities plant.

The rate of removal of liquid water from the environment would amount to about 45 ℓ/sec (740 gpm). Of this amount, about 35 ℓ/sec is required for cooling tower makeup and the remainder for various process uses.

3.2.3.5 Fuel Reprocessing Plant Emissions

Emissions from the reference FRP are characterized in Table 3.2.10.

TABLE 3.2.10. Fuel Reprocessing Plant Emissions

Emission	Description	Annual Quality	
Gaseous	Ventilation air	Air	8×10^9 kg
	Vaporized excess water	H_2O	2×10^7 kg
	Combustion products	$NO + NO_2$	1×10^5 kg
		SO_2	1×10^5 kg
		CO	1×10^3 kg
		CO_2	4×10^7 kg
Cooling tower water:	Evaporated, T = 38°C	H_2O	9×10^8 kg
	Drift, T = 38°C	H_2O	4×10^7 kg
	Blowdown, T = 27°C	H_2O	2×10^8 kg
Liquids	Nonradioactive liquid effluents	H_2O	1×10^8 kg
		SO_4^{-2}	3×10^3 kg
		NO_3^-	3×10^3 kg
		Cl^-	5×10^3 kg
		$Na^+ + K^+$	1×10^4 kg
Other	Heat	8×10^5 MW-hr (3×10^{12} BTU)	

3.2.3.6 Fuel Reprocessing Plant Costs

The capital cost estimate for the FRP is shown in Table 3.2.11 expressed in mid-year 1976 dollars. The estimate is representative of a typical plant built at a midwestern site and licensed for operation in the 1980-90 time period.

The total capital cost includes all plant-related costs incurred from the start of engineering to the initiation of commercial operation with the exception of working capital and the specific exclusions stated below. A complete description of the cost estimate bases, assumptions, and definitions is given in Section 3.8.

Two general categories are excluded from this estimate. These are:

- costs of the onsite radioactive waste management systems and facilities specified at the beginning of Section 3.2.3.3
- costs of shipping casks and other radioactive material containers primarily intended for use offsite.

TABLE 3.2.11. Fuel Reprocessing Plant Capital Cost Estimate

Cost Element	Man-hours, 1000s		Costs, 1000s of Mid-1976 Dollars		
	Nonmanual	Manual	Material	Labor	Total
Major equipment		600	78,000	8,000	86,000
Buildings and structures		3,600	39,000	43,000	82,000
Bulk materials		4,100	58,000	49,000	107,000
Site improvements		600	9,000	7,000	16,000
Subtotal of direct site construction costs		8,900	184,000	107,000	291,000
Indirect site construction costs	2,300	1,800	41,000	53,000	94,000
Total field cost	2,300	10,700	225,000	160,000	385,000
Architect-engineer services					65,000
Subtotal					450,000
Owner's cost					150,000
Total facility cost					600,000
Estimate accuracy range					±20%

The accuracy range indicated in Table 3.2.11 reflects uncertainties in the scope of engineering required to provide a fully functional plant based on the described technology, and uncertainties in the prices and quantities of labor, materials, and equipment. A contingency covering these and similar factors has been included in the base estimate. With this contingency included, there is an approximately equal likelihood of the indicated cost being overrun or underrun.

Operating costs of the reference FRP as described here are estimated to be in the range of $30 to $50 million per year.

3.2.3.7 Fuel Reprocessing Plant Construction Requirements

Many factors relating to site preparation and construction of the FRP as described in this section may have some impact on the environment, the local economy, and the natural resources of the surrounding area. The information that follows provides a basis for evaluating the impact of construction activities.

Project Schedules and Construction Manpower. The estimated schedule for engineering, procurement and construction of the FRP is shown in Figure 3.2.13. The construction labor force size, composition and schedule are shown in Figure 3.2.14.

Distribution of Onsite and Offsite Costs. Onsite costs are those for all construction, materials, and services provided at the site of the FRP while offsite costs are those for all services provided, equipment fabricated or assembled, and materials purchased elsewhere. The distribution of total costs in these categories is shown below:

Onsite costs	$175,000,000
Offsite costs	425,000,000
Total	$600,000,000

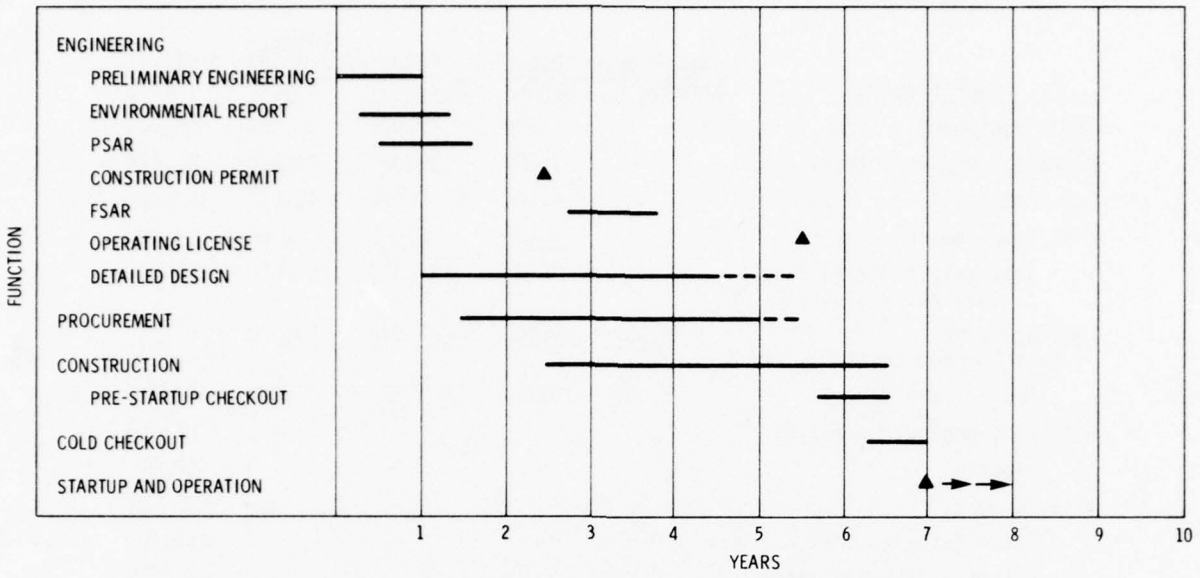

FIGURE 3.2.13. Fuel Reprocessing Plant Engineering, Procurement and Construction Schedule

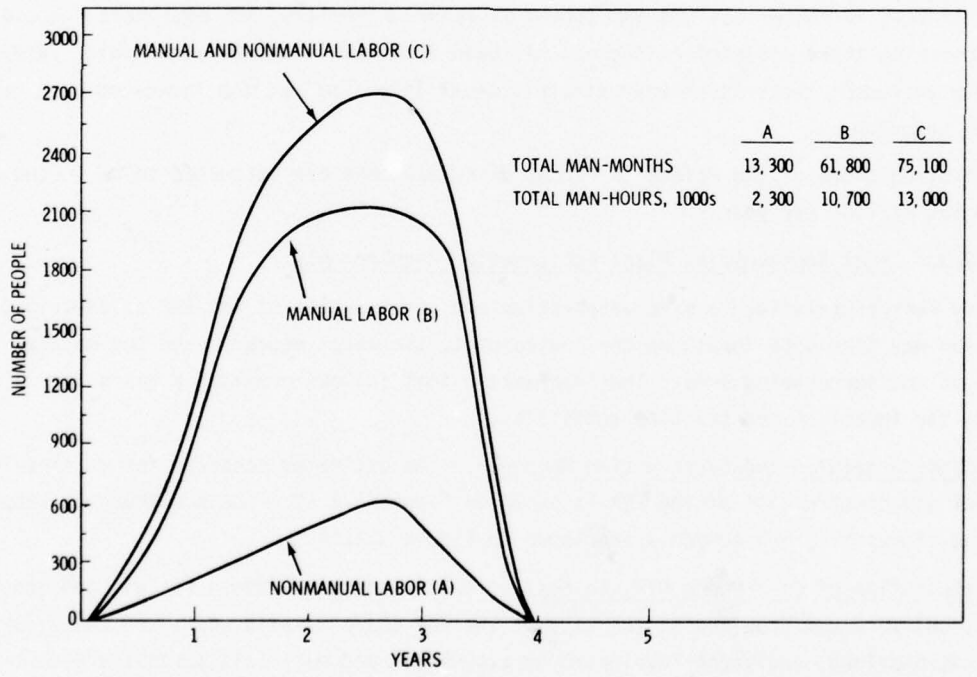

FIGURE 3.2.14. Fuel Reprocessing Plant Construction Labor Force Schedule

Temporary Construction Facilities. Temporary facilities include:

- temporary field offices
- construction warehouse and tool room
- shops for piping, electrical, instrumentation and similar crafts
- construction parking areas
- construction storage areas.

Construction Equipment. A large complement of construction equipment is required. Major items include:

- three 150-ton crawler cranes of the Manitowoc-type or similar
- concrete batch plant of 76 m^3/hr (100 yd^3/hr) capacity. A source of sand and gravel to feed this plant must also be developed.

Site Requirements. The FRP site is about 2400 ha (6000 acres), approximately 40 ha (100 acres) of which are required for facility installations. Approximately 60 additional hectares (150 acres) are required for construction storage, work yards, temporary buildings, and labor parking.

Water. About 1.8×10^5 m^3 (4.7×10^7 gal) of water are required during the construction period.

Construction Materials. Major material requirements for facility construction are:

Concrete	120,000 m^3	(150,000 yd^3)
Steel	25,700 MT	(27,200 tons)
Copper (mainly wire and cable)	145 MT	(160 tons)
Zinc	9 MT	(10 tons)
Aluminum	220 MT	(240 tons)
Lumber	4,700 m^3	(2,000 MFBM)

Energy. Energy resources used during construction will be:

Propane	1.5×10^6 ℓ	(400,000 gal)
Diesel fuel	7.6×10^6 ℓ	(2,000,000 gal)
Gasoline	9.8×10^6 ℓ	(2,600,000 gal)
Electricity		
Peak demand	3,000 kW	
Total energy	7.4×10^5 kWh	

Transportation Requirements. A railroad spur track about 4.8 km (3 miles) long must be brought into the construction site. This track must be routed and constructed to suit the final plant layout for use in bringing in the spent fuel casks mounted on rail cars. A 4.8 km (3 mile), two-lane paved road must also be constructed to the job site. This will be required for worker's traffic and for material and equipment deliveries by truck. A future use will be for the hauling of spent fuel casks by truck.

3.2.3.8 Effects of Fuel Cycle Options

Use of the reference FRP assumes reprocessing of spent LWR fuel and recycling of purified uranium and plutonium. Other uranium and plutonium recycle options, such as "coprocessing" (leaving the plutonium mixed with at least part of the uranium) and "partial decontamination" (leaving more of the fission products with the plutonium), could be accomplished in a similar but possibly slightly modified plant. The following fuel cycle modes have also been assessed as they relate to an FRP.

No Recycle. Eliminating fuel recycle would do away with the need for an FRP.

Uranium Recycle Only, with Plutonium to a Repository. This alternative would have little effect on the FRP except for possibly reducing the required number of plutonium purification cycles.

Uranium Recycle Only, with Plutonium to High-level Waste. This alternative would require modification of some process stream flows, possibly require some additional and/or modified interim solution storage capability, and would eliminate some plutonium purification and conversion operations.

3.2.4 Mixed Oxide Fuel Fabrication Plant

The reference mixed oxide fuel fabrication plant (MOX FFP) and its construction and operating requirements are described here to provide a basis for assessing the requirements for the related radioactive waste management functions. The purpose of the MOX FFP is to utilize plutonium recovered in an FRP in fuel to be recycled through an LWR. At the MOX FFP, plutonium dioxide is mixed with natural uranium dioxide to produce a fuel containing about 3% fissile plutonium (about 5% total plutonium).

3.2.4.1 Selection of the Reference Mixed Oxide Fuel Fabrication Plant

The reference MOX FFP incorporates a process that mixes UO_2 and PuO_2 powders, forming the mixture into dense fuel pellets, and sealing the fuel pellets within Zircaloy cladding tubes. The reference plant employs this process since it appears to be the one favored by industry after examining other methods, as evidenced by commercial designs and the construction of small-scale fuel fabrication plants.

3.2.4.2 Mixed Oxide Fuel Fabrication Plant Process

The reference MOX FFP receives UO_2 and PuO_2 powders, zirconium alloy tubes, hold-down springs, and end plugs from offsite producers. The plant produces sealed fuel rods ready for insertion into fuel assemblies. The plant capacity is 400 MTHM of mixed oxide fuel per year. This size facility approximately matches the plutonium recovery rate from the reference FRP.

A schematic diagram showing the major process operations is shown as Figure 3.2.15. This figure also illustrates the sources of liquid and gaseous wastes that result from operation of the MOX FFP.

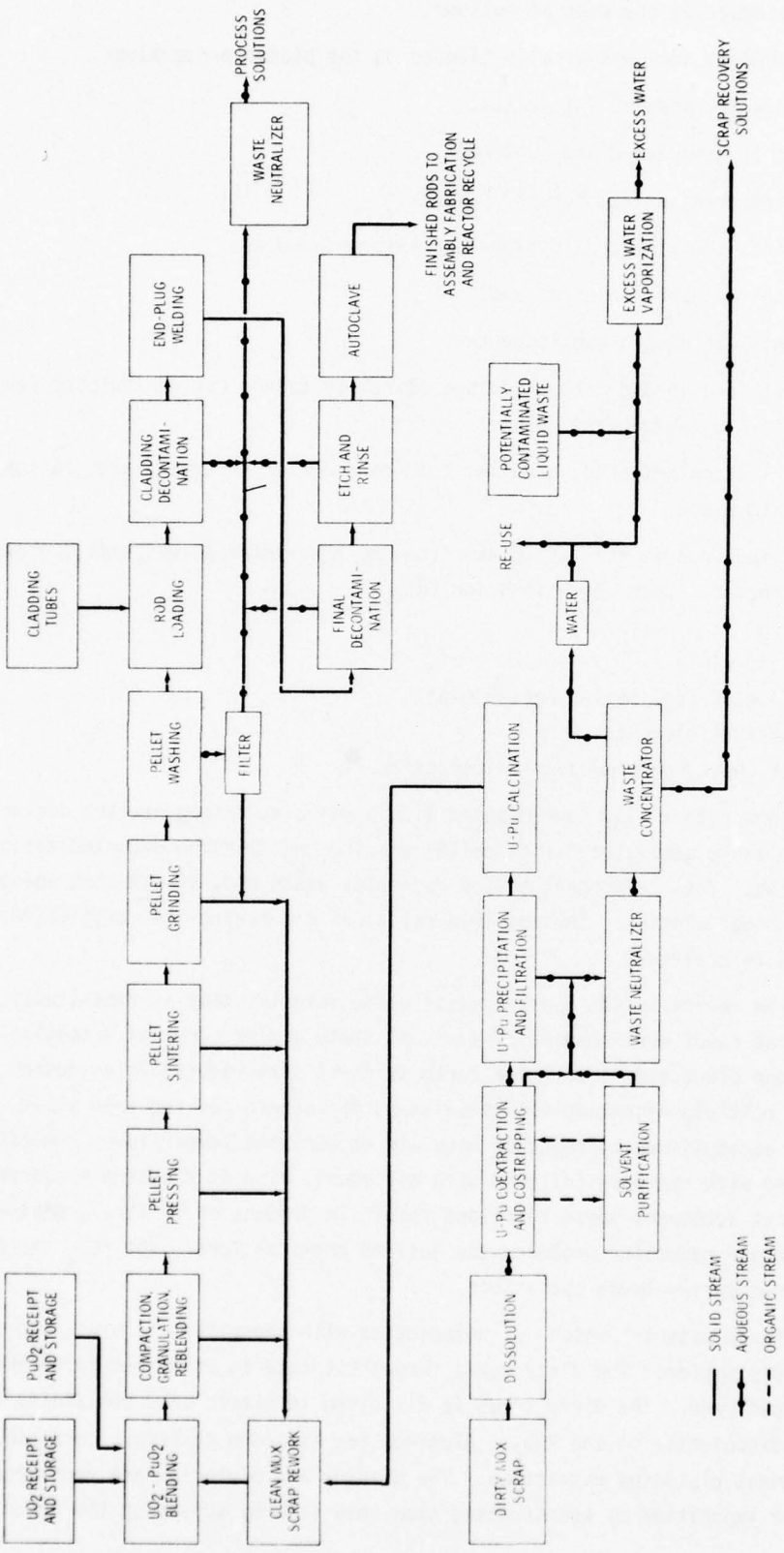

FIGURE 3.2.15. Mixed Oxide Fuel Fabrication Plant Process Flow Diagram

The fuel rod production proceeds as follows:

- PuO_2 and UO_2 powders are mechanically blended in the proper proportions.
- The mixed oxides are pressed into cakes.
- The oxide cake is granulated and reblended.
- The granules are pressed into "green" pellets.
- The green pellets are sintered in high-temperature furnaces.
- The sintered pellets are finish ground.
- Finished pellets are washed and inspected.
- Finished pellets are loaded into cladding, (Zircaloy tubing cut to finished length and with bottom end-plug welded into place).
- The cladding is decontaminated, a pellet hold-down device inserted, and the top end-plug welded into place.

The completed fuel rod is then withdrawn from the restricted access area and additional operations are performed. Such operations include:

- clad degreasing
- clad etching and rinse
- x-ray inspection of fuel pellet arrangement
- autoclaving of fuel elements
- fuel-clad leak check and dimensional inspection.

Contaminated and potentially contaminated liquid waste streams generated during the processing include those generated during pellet washing and cladding decontamination, etching, and rinsing. The cladding-etch step generates waste HNO_3-HF-H_2O etch solution and $Al(NO_3)_3$-H_2O etch-stop solution. The combined solutions are neutralized with $Ca(OH)_2$ in preparation for waste treatment.

The clean scrap rework system handles mixed oxide material that is essentially free of impurities, but that needs modification of physical state and/or chemical oxidation state before rejoining the production line. The scrap is first size-reduced in a crusher and is then subjected to multiple oxidation-reduction steps to achieve the required state. Oxidation of UO_2 to U_3O_8 is accomplished by reaction with air at elevated temperature; reduction back to UO_2 is accomplished with hydrogen (diluted with nitrogen), also at elevated temperature. The density changes that accompany these reactions result in production of finely-divided powder, and the final hydrogen reduction produces the desired chemical form. The PuO_2 remains chemically unaltered during these operations.

Mixed oxide scrap material which is contaminated with impurities is processed in the dirty scrap recovery system. The dirty scrap throughput rate is estimated to be about 2% of the plant throughput rate. The dirty scrap is dissolved in nitric acid containing fluoride ion to assist in dissolution of the PuO_2. Aluminum ion is added to reduce corrosion and to facilitate subsequent plutonium extraction. The uranium and plutonium are recovered and separated from the impurities by coextracting them into TBP, by scrubbing the organic extract

with a nitric acid solution, and by costripping them into dilute nitric acid. The uranium and plutonium are then coprecipitated with ammonia, and calcined to oxides before being returned to the clean scrap rework system. The aqueous streams from the coextraction and scrubbing operations and from solvent washing are combined with the supernatant solution from the precipitation step and are neutralized with calcium hydroxide prior to concentration. The concentrator bottoms stream is a waste solution to be treated. The concentrator overheads stream (water) is reused in the process or is vaporized (along with other potentially contaminated liquid wastes) for discharge to the environment.

Vaporized excess water is the major process off-gas stream from the MOX FFP. This stream is mixed with the large volume building ventilation exhaust air and discharged up the stack. The only other source is the off-gas from the dirty scrap dissolver; this small stream is scrubbed to remove the bulk of the nitrogen oxides before it is mixed with the ventilation air stream for discharge. Since gaseous radioactive materials are not present in the MOX FFP, the gaseous effluent treatment system is designed solely to retain solid particles (Section 4.9).

3.2.4.3 Mixed Oxide Fuel Fabrication Plant Description

As with the other primary facilities, the description given here of the MOX FFP is a general one providing a background for the following detailed discussions of associated waste management facilities. These waste management facilities are not considered in the information presented in this section on the MOX FFP. Such facilities sharing the MOX FFP site, but not directly involved in the fuel fabrication operations, are:

- failed equipment and noncombustible waste treatment (Section 4.3)
- compactable and combustible waste treatment (Section 4.4)
- wet waste and particulate solids immobilization (Section 4.7).

Figure 3.2.16 is a plot plan of the MOX FFP. The plant is built on a 400-ha (1000-acre) site, fenced with posted agricultural-type fencing. The main plant facilities are located near the center of the site within a protected security area of about 3 ha (8 acres). All systems and services involved in the handling of PuO_2 powders or MOX pellets are housed in Category I structures, as are essential emergency features of the plant. These structures are indicated by heavy outlines on the plot plan. The main plant is divided into the following functional areas characterized by the processes or operations carried on in each.

Production Materials Receiving and Preparation Area. In this area, the nonsensitive production materials (principally UO_2, tubes, and hardware) are received, inspected, and prepared for use in the fuel manufacturing process. This area is divided into two separate, noncontiguous subareas--one devoted to UO_2 receipt and temporary storage, and the other to tube, spring, and fuel rod end-plug inspection and preassembly. Materials of this type are received onsite in standard containers conveyed by common-carrier trucks.

PuO_2 Receiving and Preparation Area. Plutonium dioxide powder is received in standard DOT specification 6M containers or advanced design containers for shipment of recycle plutonium. All shipments are expected to be received in safeguards-equipped vehicles. Shipments of PuO_2 are received through a special entrance to the plant which is used for no other

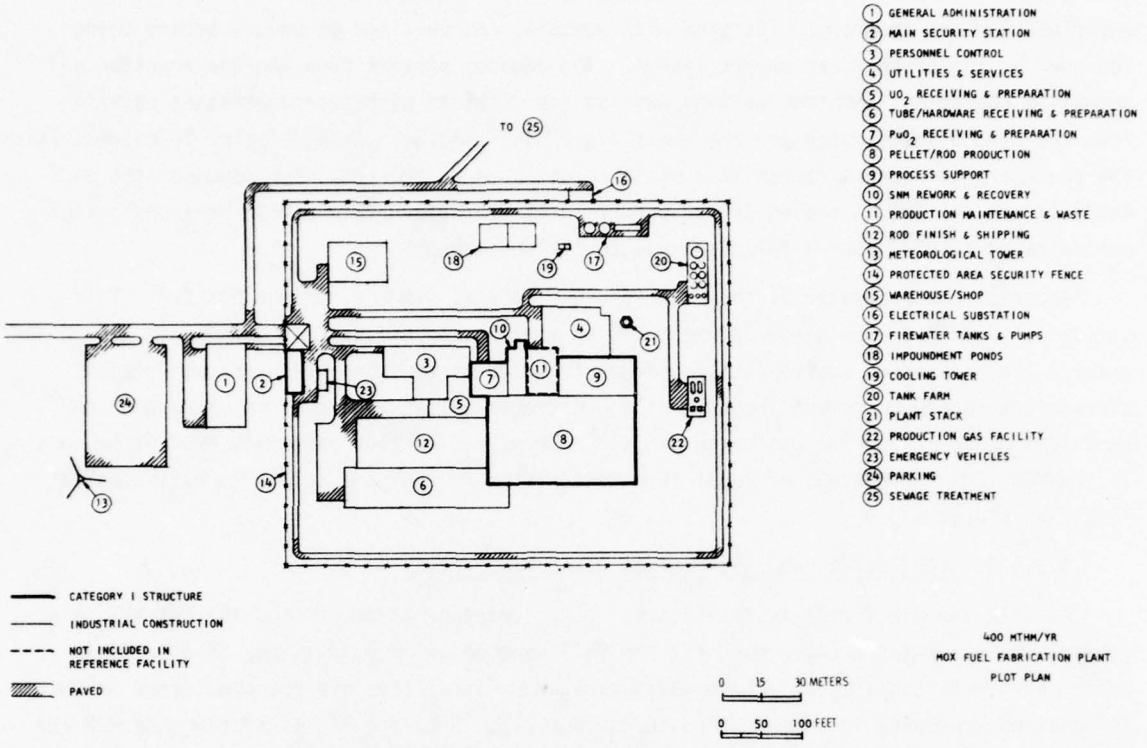

FIGURE 3.2.16. Mixed Oxide Fuel Fabrication Plant

purpose. After immediate accountability checks and container surface inspection, the PuO_2 is temporarily stored in its shipping container. At the time of unloading, the shipping container is moved into a remotely-operated shielded enclosure where the PuO_2 is removed from its inner container and stored in bins while the components of the container are cleaned for return or discard.

Pellet/Rod Production Area. The operations associated with blending UO_2 and PuO_2 powders, slugging, granulation, pellet forming, sintering, inspection, rod loading and seal welding are carried out in this area. Two parallel, shielded restricted access areas (RAAs) are provided in which all of the routine production operations are carried out remotely through extensive use of automation. The RAAs are surrounded by limited access areas (LAAs) in which personnel may perform a limited amount of maintenance on equipment or nonroutine production operations through glove ports or by other means.

The LAAs are, in turn, surrounded by normal access areas (NAAs) in which the routine production operations are controlled and monitored by plant operating personnel.

Special Nuclear Material (SNM) Rework and Recovery Area. The main functions carried out in this section of the plant are the rework of clean MOX scrap and the processing of the dirty MOX scrap.

Process Support Area. General facilities and systems that support production operations are housed in this area. These include analytical laboratories, and emergency power and control facilities.

Personnel Control and Technical Office Area. This area provides space for personnel facilities such as lockers, change rooms, health physics and medical facilities. Technical and production control office space is also provided in this area.

Utilities and Services Area. Normal, nonsensitive utilities are housed in this area, including electrical switchgear, air compressors, and other equipment.

Main Security Station. The security station provides all facilities necessary for complete personnel and vehicle identification and inspection. It also acts as a central monitoring point for perimeter and yard surveillance and is connected to security check points within the main plant. It is constructed and equipped to resist armed assault.

Miscellaneous Facilities. Yard facilities include a tank farm, an area for the storage, mixing and metering of production gases used in pellet fabrication and rework, fire water tanks and pumps, impoundment ponds, a combination warehouse and shop and similar facilities. The general administrative offices, meteorological tower, and secondary sewage treatment facilities are outside of the protected security area.

Facility characteristics of special importance to environmental considerations are those of the exhaust stack and the cooling tower. Tables 3.2.12 and 3.2.13 list these characteristics.

TABLE 3.2.12. Mixed Oxide Fuel Fabrication Plant Cooling Tower Data

Heat load	= 3 MW
Water flow:	
Circulating	= 62 ℓ/sec @ 27-38°C
Evaporated	= 1.2 ℓ/sec @ 38°C
In drift	= 0.006 ℓ/sec @ 38°C
In blowdown	= 0.22 ℓ/sec @ 27°C
Makeup	= 1.4 ℓ/sec

Note: 1 ℓ/sec = 15.8 gpm

TABLE 3.2.13. Mixed Oxide Fuel Fabrication Plant Exhaust Stack Data

Exhaust flow	= 15 m^3/sec
Exhaust velocity	= 15 m/sec
Release height	= 20 m

Shielding and Remote Handling Equipment. Special shielding and remote handling equipment are required for much of the facility and for many of the items of equipment. Required shielding is not substantial. Special handling provisions are mainly for the physical containment of the radioactive materials. Extensive use is made of automated, remote operations, and of operations through glove ports. Additional special handling considerations are required by criticality safety considerations.

3.2.4.4 Mixed Oxide Fuel Fabrication Plant Operating and Maintenance Requirements

Operation of the reference MOX FFP mainline process is assumed to be continuous (24 hours per day, 300 days per year). The dirty scrap recovery process is assumed to operate in relatively brief campaigns (days to weeks) when the accumulation of scrap reaches an appropriate level (a few times a year).

Unusual maintenance requirements do exist since some maintenance must be performed on contaminated facilities and equipment. Special facilities and equipment are provided to allow such maintenance to be performed.

Utilities. The electrical power consumption for the MOX FFP would be about 3 MW. The rate of removal of liquid water from the environment would amount to about 2.5 ℓ/sec (40 gal/min). About half of this water is for cooling tower makeup and the remainder for various process uses.

Staffing. The reference MOX plant would require a staff of about 300 people.

3.2.4.5 Mixed Oxide Fuel Fabrication Plant Emissions

Emissions from the reference MOX FFP are characterized in Table 3.2.14.

TABLE 3.2.14. Mixed Oxide Fuel Fabrication Plant Emissions

Emission	Description	Annual Quantity	
Gaseous	Ventilation air	Air	5×10^8 kg
	Vaporized excess water	H_2O	5×10^5 kg
	Process releases	N_2	1×10^6 kg
		H_2	4×10^3 kg
		$NO + NO_2$	1×10^2 kg
		NH_3	5×10^1 kg
Cooling tower water:	Evaporated, T = 38°C	H_2O	4×10^7 kg
	Drift, T = 38°C	H_2O	1×10^5 kg
	Blowdown, T = 27°C	H_2O	7×10^6 kg
Liquids	Nonradioactive liquid effluents	H_2O	2×10^7 kg
		PO_4^{-3}	3×10^1 kg
		NO_3^-	1×10^3 kg
		Na^+	4×10^2 kg
Other	Heat	3×10^4 MW-hr (1×10^{11} BTU)	

3.2.4.6 Mixed Oxide Fuel Fabrication Plant Costs

The capital cost estimate for the MOX FFP is shown in Table 3.2.15 expressed in mid-1976 dollars. The estimate is based on an automated, semi-remotely maintained facility

and reflects anticipated costs associated with stringent health and safety measures, continuous special nuclear material (SNM) accountability, and an advanced safeguards/security program.

TABLE 3.2.15. Mixed Oxide Fuel Fabrication Plant Capital Cost Estimate

Cost Element	Man-hours, 1000s		Costs, 1000s of Mid-1976 Dollars		
	Nonmanual	Manual	Material	Labor	Total
Major equipment		300	18,000	4,000	22,000
Buildings and structures		1,200	5,000	15,000	20,000
Bulk materials		1,200	10,500	15,500	26,000
Site improvements		100	500	1,500	2,000
Subtotal of direct site construction costs		2,800	34,000	36,000	70,000
Indirect site construction costs	700	500	13,000	18,000	31,000
Total field cost	700	3,300	47,000	54,000	101,000
Architect-engineering services					20,000
Subtotal					121,000
Owner's cost					39,000
Total facility cost					160,000
Estimate accuracy range					±30%

The total capital cost includes all plant-related costs incurred from the start of engineering to the initiation of commercial operation with the exception of working capital and the specific exclusions stated below. A complete description of the cost estimate bases, assumptions, and definitions is given in Section 3.8. Two general categories of costs are excluded from this estimate. These are:

- costs of onsite radioactive waste management systems and facilities
- costs of plutonium and waste shipping containers and transporters intended for use primarily offsite.

The estimate accuracy range reflects uncertainties in the engineering scope required to provide a fully functional plant based on the technology described and in the quantities and pricing for labor, materials, and equipment. A contingency covering these and similar factors has been included in the base estimate. With the contingency included, there is an approximately equal likelihood of the indicated cost being overrun or underrun.

Operating costs of the reference MOX FFP are estimated to be in the range of $30 million to $60 million per year.

3.2.4.7 Mixed Oxide Fuel Fabrication Plant Construction Requirements

Many factors relating to site preparation and construction of the MOX FFP as described in this section may have some impact on the environment, the local economy, and the natural resources of the surrounding area. The information that follows provides a basis for evaluating the impact of construction activities.

Project Schedules and Construction Manpower. The estimated schedule for engineering procurement and construction is shown on Figure 3.2.17. The construction labor force size, composition and schedule are shown in Figure 3.2.18.

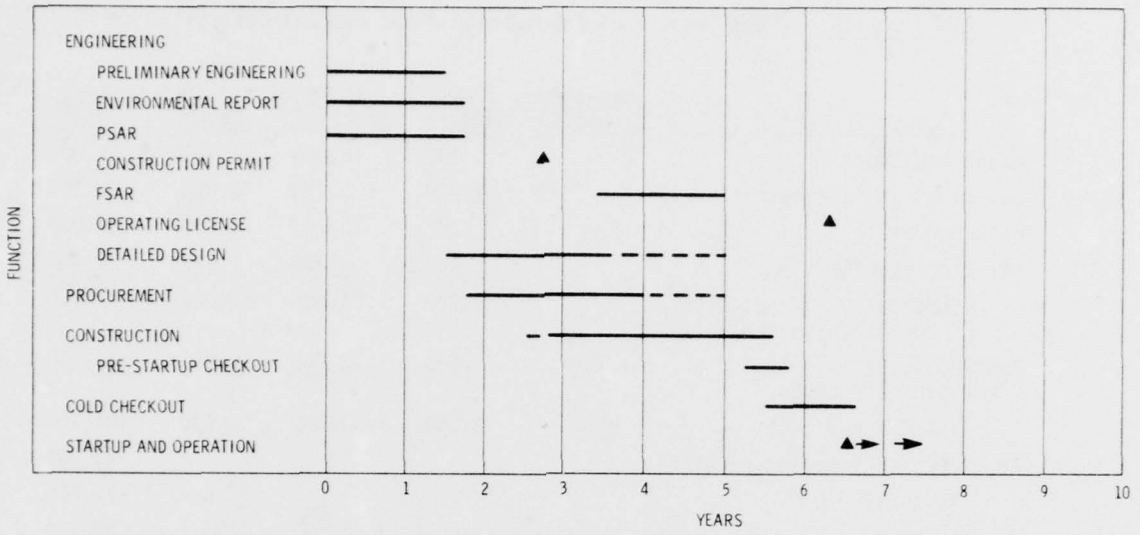

FIGURE 3.2.17. Mixed Oxide Fuel Fabrication Plant Engineering, Procurement and Construction Schedule

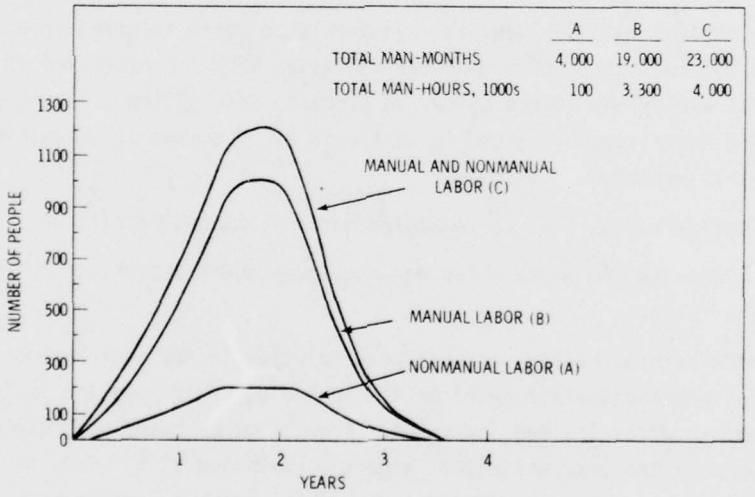

FIGURE 3.2.18. Mixed Oxide Fuel Fabrication Plant Construction Labor Force Schedule

Distribution of Onsite and Offsite Costs. Onsite costs are those for all construction materials and services provided at the site, while offsite costs are those for all services provided, equipment fabricated or assembled, and materials purchased elsewhere. The distribution of total costs in these categories is shown below:

Onsite costs	$ 62,000,000
Offsite costs	98,000,000
Total	$160,000,000

Temporary Construction Facilities. Ten ha (25 acres) will be required for temporary construction facilities, to include:

- temporary field offices
- construction warehouse and tool room
- shops for piping, electrical, instrumentation and similar crafts
- construction parking areas
- construction storage areas.

Construction Equipment. Equipment will be typical of normal industrial-type projects. An onsite concrete batch plate will probably not be required.

Site Requirements. The MOX FFP requires a 400-ha (1000-acre) site, with 6 ha (15 acres) devoted to facilities.

Water. About 4.9×10^4 m^3 (1.3×10^7 gal) of water will be used during the construction period.

Construction Materials. Major material requirements for facility construction are:

Concrete	11,000 m^3	(15,000 yd^3)
Steel	6,400 MT	(7,000 tons)
Copper (mainly wire and cable)	54 MT	(60 tons)
Zinc	9 MT	(10 tons)
Aluminum	4 MT	(5 tons)
Lumber	710 m^3	(300,000 board feet)

Energy. Energy resources used during construction will be:

Propane	380,000 ℓ	(100,000 gal)
Diesel fuel	4,500,000 ℓ	(1,200,000 gal)
Gasoline	3,000,000 ℓ	(800,000 gal)
Electricity		
Peak demand	1,000 kW	
Total consumption	3.2×10^6 kWh	

Transportation Requirements. A new, two-lane paved road approximately 1.6 km (1 mile) long will be required to support facility construction and subsequent plant operation. Rail access is not required.

3.2.4.8 Effects of Fuel Cycle Options

Use of the reference MOX FFP assumes reprocessing of spent LWR fuel and recycling both the recovered uranium and plutonium. The MOX FFP would not be needed if the alternative fuel cycle modes of no recycle or of uranium-only recycle were implemented.

3.3 WASTE DESCRIPTIONS AND CLASSIFICATIONS

3.3 WASTE DESCRIPTIONS AND CLASSIFICATIONS

For the purposes of this report, radioactive wastes are defined as all gases, liquids and solids that by virtue of their use in any part of the postfission light water reactor (LWR) fuel cycle, are actually or potentially contaminated with radioactivity and have been classified worn out, defective, or of no further use. This section characterizes primary wastes from the postfission commercial LWR fuel cycle, i.e., wastes from the operation of nuclear power plants, independent spent fuel storage basins, fuel reprocessing plants, and mixed oxide fuel fabrication plants. Nuclear power plant gaseous wastes are not included in the scope of this report, however, since they are extensively described and evaluated in individual nuclear plant environmental impact statements. Additional subsidiary, or secondary, wastes resulting from treatment and handling of the primary wastes are described in Section 3.5. Wastes that do not require management until the time of plant decommissioning are described in Section 8.

Characterization of primary wastes requires definition of the following:

- composition of the waste
- rate of waste generation
- associated activity.

In addition to defining these properties, it is helpful to categorize the wide variety of primary wastes generated. Many methods of classifying radioactive wastes are in use, based on parameters such as the kind of radioactivity present, the amount of radioactivity present, the untreated physical form, and the treated physical form. For this report, which analyzes the effects of waste treatment alternatives, it is convenient to classify the primary wastes into categories based on their treatment requirements as shown in Table 3.3.1.

Wastes in the first four categories listed below are generated to some degree in almost any facility in which radioactive materials are processed, treated or handled; as is evident from the list, the remaining four waste categories are specific to certain fuel cycles.

Packaged treated wastes are classified by surface radiation levels to define handling, shipping and interim storage requirements. Low-level wastes (LLW) are defined here as having surface radiation levels less than 200 mrem/hr and require little or no shielding during handling, shipping and interim storage. Intermediate-level wastes (ILW) are defined here as having surface radiation levels greater than 200 mrem/hr and always require shielding during handling, shipping and interim storage. The nomenclature, high-level waste (HLW), is reserved for one specific waste, the fission product stream from the first extraction cycle in a reprocessing plant. HLW is self-heating and requires shielding plus provisions for cooling during interim storage. Spent fuel assemblies have the same interim storage requirements as HLW.

Final disposal requirements dictate that there be one further classification of packaged treated wastes. Non-transuranic wastes can be disposed of in surface burial grounds; transuranic (TRU) wastes are assumed to require disposal in geologic repositories. A proposed rulemaking would consign to licensed burial grounds wastes which are contaminated with no more than 10 nCi of TRU/g. Thus wastes with greater than 10 nCi of TRU/g become TRU wastes. In practice, because this concentration is difficult to detect, all wastes from locations that might cause contamination levels above 10 nCi TRU/g of waste are often treated as TRU-suspect and combined

TABLE 3.3.1. Categorization of Wastes According to Treatment Requirements

Waste Category	Applicable Alternative Fuel Cycle	Treatment Requirements
Gaseous	All	Reduce activity to approved levels by removing particulate and volatile radioisotopes.
Compactable trash and combustible wastes	All	Reduce volume. Reduce combustion hazard. Package in approved container for shipment and disposal.
Concentrated liquids, wet wastes, and particulate solids	All	Convert to approved solid form. Package in approved container for shipment and disposal.
Failed equipment and noncompactable, noncombustible wastes	All	Reduce activity level by decontamination. Reduce volume by dismantlement, cutting, etc. Package in approved container for shipment and disposal.
Spent fuel assemblies	Once-through	Package in approved container for interim storage and disposal.
High-level liquid waste	Uranium recycle and uranium-plutonium recycle	Convert to approved solid form. Package in approved container for shipment and disposal.
Fuel residue (hulls and assembly hardware)	Uranium recycle and uranium-plutonium recycle	Reduce volume, if practical. Reduce activity level, if practical. Package in approved container for shipment and disposal.
Plutonium oxide	Uranium recycle	Package in approved container for interim storage and potential disposal.

with the known TRU wastes. That is the operational philosophy assumed for this report. Thus all wastes from nuclear power plants and independent spent fuel storage basins are non-TRU wastes. All wastes from mixed oxide fuel fabrication plants are assumed to be TRU wastes. The classification of fuel reprocessing plant wastes as TRU or non-TRU is dependent on the source of the primary wastes within the plant.

For the sake of completeness, both the non-TRU and TRU wastes from the postfission LWR fuel cycle are described in this section, even though non-TRU waste management, except for post-fission gaseous wastes, is outside the scope of this report. Only TRU and post-fission gaseous waste treatment, storage, transportation, and disposal is described in the remainder of this report.

As a basis for characterizing the primary wastes discussed in this report, a reference spent fuel was defined for each of three fuel cycle modes: the once-through, the uranium-only recycle, and the uranium and plutonium recycle fuel cycle modes. These reference fuels comprise the mixture of spent fuels projected to be discharged from pressurized water reactors (PWRs) and (BWRs) in the year 2000, as shown in Table 3.3.2.

TABLE 3.3.2. Reference Fuel Components for Three Fuel Cycle Modes

Reactor Discharge[a]	Contribution of Specific Reactor Discharge to Total Reference Fuel, %			Initial Enrichment, Weight Percent Fissile
	Once-Through	Uranium-Only Recycle	Uranium-Plutonium Recycle	
PWR 1st discharge, 15,000 MWd/MTHM	2.9	2.9	2.9	2.0
PWR 2nd discharge, 25,000 MWd/MTHM	3.0	3.0	3.0	2.6
PWR 3rd discharge, 33,000 MWd/MTHM	3.4	3.4	3.4	3.1
PWR equilibrium discharge, 33,000 MWd/MTHM	52.8	44.9	35.9	3.2
PWR recycle U discharge, 33,000 MWd/MTHM		7.9	5.3	3.6 to 4.7
PWR recycle Pu discharge, 33,000 MWd/MTHM			11.5	3.5 to 4.8
BWR 1st discharge, 10,000 MWd/MTHM	1.8	1.8	1.8	1.7
BWR 2nd discharge, 17,000 MWd/MTHM	1.9	1.9	1.9	1.9
BWR 3rd discharge, 20,000 MWd/MTHM	2.1	2.1	2.1	1.9
BWR 4th discharge, 21,000 MWd/MTHM	2.3	2.3	2.3	1.9
BWR equilibrium discharge, 27,000 MWd/MTHM	29.9	25.9	20.6	2.6
BWR recycle U discharge, 27,000 MWd/MTHM		3.9	2.8	2.9 to 3.8
BWR recycle Pu discharge, 27,000 MWd/MTHM			6.5	2.8 to 4.0

a. Residence time for PWR 1st, 2nd, 3rd, equilbrium and recycle discharges = 1, 2, 3, 3 and 3 years respectively. Residence time for BWR 1st, 2nd, 3rd, 4th equilibrium and recycle discharges = 1, 2, 3, 4 and 4 years, respectively. Average exposure for all three fuel cycle modes = 29,300 MWd/MTHM.

Table 3.3.3 summarizes the estimated nonvolatile activity distribution in primary TRU wastes from the reference fuel reprocessing plant (FRP) and mixed oxide fuel fabrication plant. The values presented in Table 3.3.3 are based on the detailed characterizations of the reference primary wastes. Except for ^{3}H, nearly all of the volatile fission products are contained in the dissolver off gas (100% of the ^{14}C and ^{85}Kr and 99% of the ^{129}I). Most of the ^{3}H (72%) is contained in the effluent from the FRP excess water vaporizer with the remainder distributed 5% to the DOG, 15% to spent fuel hulls, and 8% to the HLLW.

The primary wastes, as defined for this report, are described in two sets of tables given at the end of this section. The first set of tables (Tables 3.3.4-3.3.18) are the reference radionuclide inventory tables. They show the full spectrum of radioisotopes, including activation products, fission products, and actinides in each fuel cycle, as a function of time. The

TABLE 3.3.3. Nonvolatile Activity Distribution in Fuel Reprocessing Plant and Mixed Oxide Fuel Fabrication Plant Primary Wastes

	Percent of Plant Input Activity Appearing in Primary Wastes					
	Nonvolatile Fission Products			Actinides		
	Zr-Nb	Ru	Others	U	Pu	Others
Fuel Reprocessing Plant						
To cladding	0.05	0.05	0.05	0.05	0.05	0.05
To high-level waste	>99.84	>99.84	>99.94	0.5	0.5	>99.94
To other solid and liquid wastes	0.1	0.1	0.001	0.1	0.1	0.001
To PuO_2 conversion wastes	neg.[a]	neg.	neg.	neg.	0.27	neg.
To UF_6 plant wastes	neg.	neg.	neg.	0.29	neg.	neg.
Mixed Oxide Fuel Fabrication Plant						
To solid and liquid wastes	0	0	0	0.12	0.12	neg.
Total	100	100	100	1.06	1.04	100

a. negligible

radionuclide inventories are given either per MTHM or per GWe-yr. The values are a weighted average for the year 2000, assuming the installed nuclear power plants in each of the fuel cycles exist in an electrical generating capacity ratio of 2 PWRs:1 BWR. The average fuel exposure in the year 2000 is estimated to be 29,300 MWd/MTHM. This value was determined by assuming an installed nuclear power capacity of 400 GWe and by projecting the mixture of nuclear power plant cores that would be discharged in the year 2000; this determination takes into account that some nuclear power plants would be discharging various startup cores whereas others have reached equilibrium operation, as shown in Table 3.3.2.

The reference fuels described in Table 3.3.2 are the fundamental source of activity in almost all components of the postfission LWR fuel cycle as defined for this report. The radionuclide composition of the reference fuels was developed from a series of ORIGEN code[1] runs weighted in the proportions shown in Table 3.3.2. The activity in any portion of the fuel cycle is proportional to the equivalent throughput of fuel in that component of the fuel cycle; e.g., the fission product activity passing through a 2000-MT/yr reprocessing plant processing uranium-only recycle fuel 1.5 years out-of-reactor is $2000 \times 1.4 \times 10^6$ (from Table 3.3.8) = 2.8×10^9 Ci/yr. If the fuel were 10 years out-of-reactor when processed, the fission product activity passing through the plant would be $2000 \times 2.7 \times 10^5$ (again from Table 3.3.8) = 5.4×10^8 Ci/yr. In the year 3000, the solidified high-level waste from this one year's operation of the reprocessing plant would contain 2000 x 19 (again from Table 3.3.8) = 38,000 Ci of fission product activity.

The activity given in Table 3.3.4 for reactor and spent fuel storage basin wastes is given in terms of Ci/GWe-yr or Ci/yr, rather than Ci/MTHM. This is because the activity which is found in reactor and spent fuel storage basin wastes is not directly proportional to the throughput of spent fuel. Instead, reactor and spent fuel storage basin waste activity is a mixture comprising activated corrosion products from reactor core components and fission product isotopes that are released at differing rates from tramp uranium* and leaking fuel rods. The activities given for the fuel basin wastes in Table 3.3.4 are specific estimates for the operating schedule defined for the reference independent spent fuel storage basin and fuel reprocessing plant.

The second set of tables at the end of this section (Tables 3.3.19 - 3.3.40) are the waste characterization tables, which arrange the wastes from the four reference primary fuel cycle facilities into the categories given in Table 3.3.1. The information given for each primary waste stream includes:

- composition (wt%)
- density
- generation rate (vol/MTHM or GWe-yr, as applicable)
- radioactivity factors.

The radioactivity in a waste stream is obtained by applying the radioactivity factor given for each individual waste stream to the pertinent reference radionuclide inventory. For instance, for a 2000-MT/yr reprocessing plant processing uranium-recycle-only spent fuel 1.5 years out-of-reactor, the plutonium-238 activity in the low-level, combustible trash from a year's operation would be $2000 \times 1 \times 10^{-9}$ (from Table 3.3.34) $\times 3.14 \times 10^{3}$ (from Table 3.3.11) $= 6.3 \times 10^{-3}$ Ci. Similarly, the activity for any isotope in any waste stream can be found by proper combination of the information in the reference radionuclide and waste characterization tables.

The values given for the properties in Tables 3.3.19 - 3.3.40 are best estimates. Sources for the estimates are given in the following text when available; in instances where no references were available, estimates were developed in this study.

3.3.1 Nuclear Power Plant Primary Wastes

The primary wastes from the reference nuclear power plant (Tables 3.3.19 - 3.3.22) represent a composite of the wastes that would result in the year 2000 from operation of LWRs for electric power production at a ratio of 2 PWRs:1 BWR. There are generic differences in the wastes generated from BWRs and PWRs; there are also substantial differences within the same reactor type, depending upon the size and age of the reactor, operating philosophy, maintenance history, etc.[2] The concern in this discussion is not, however, the various combinations of sources and flows, but rather that a description be available for every important waste form

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* Tramp uranium is minute uranium particles still adhering to the fuel cladding surfaces after fuel fabrication.

requiring treatment and disposal from both PWRs and BWRs. Therefore, the wastes include those typical of both PWRs and BWRs on a 2 PWR:1 BWR weighted average.

Wastes are produced at PWRs and BWRs by a number of routine operations: 1) water cleanup operations on reactor coolants, spent fuel basin coolants, turbine condensate, and collected liquid wastes; 2) air cleaning operations on ventilation air and all off-gases; and 3) plant maintenance and general support operations.

Sources of radioactivity in the wastes include:[3,4]

- neutron activation of structural metals and their corrosion products
- irradiation of tramp uranium on the external surfaces of fuel elements
- neutron activation of coolant components, including coolant additives
- control rods
- fission products from irradiated fuel (only in the event of cladding failure).

The isotopic distribution of radioactivity from nonvolatile fission and neutron activation products in nuclear power plant wastes is, for practical purposes, the same for all the fuel cycle modes described in Section 3.1. Therefore, the estimated radioactivity given for these wastes applies to all three fuel cycle modes.

The volatile fission products (krypton, xenon and iodine) released by cladding failure, as well as particulate solids bearing radioactivity, are treated by the gaseous radioactive waste system at the nuclear power plants. These systems are treated in detail in nuclear power plant licensing proceedings and are not considered in this report; system components that become contaminated and must be disposed of, such as filters, are described, however.

Most of the semivolatile fission product tritium released through cladding failure is expected to leave the reactor via aqueous streams[4] in vapor or liquid form; a significant fraction, however, is also expected to be borne on wet wastes as tritiated water associated with resins, slurries and sludges. The soluble, nonvolatile constituents of irradiated fuel that escape through cladding failures remain largely in the primary coolant, where their concentrations are controlled by the coolant water cleanup systems. Lesser amounts of these irradiated fuel constituents and activated corrosion products are also scavenged out of the condensate system by filtration and demineralization.

3.3.1.1 Compactable and Combustible Wastes from the Nuclear Power Plant

Compactable and combustible wastes from the reference nuclear power plant (Table 3.3.19) include high-efficiency particulate air (HEPA) filters, which become waste when they are replaced because of loss of filter efficiency or because of high activity levels. The radioactivity associated with these filters is mainly from air entrainment of dried material from coolant leaks. The particulate emission rate through the filters was estimated from calculations for PWRs[5] and BWRs.[6] Using this emission rate, and assuming a decontamination factor of 1000, the activity on the HEPA filters was calculated. The volume of HEPA filter waste was estimated from operating experience of the H. B. Robinson Plant.[7]

The volume of combustible trash was estimated from a previous description of nuclear power plant trash[8] and assumes that 80% of that total was combustible. The radioactivity of combustible trash derives from cleanup of cooling system leaks, servicing of coolant-carrying plumbing, and analytical operations. The isotopic distribution of radioactivity is based on previously published distributions of radioactivity in reactor solid wastes.[9] The total radioactivity is assumed to be 80% of the total previously cited for nuclear power plant trash.[8]

3.3.1.2 Concentrated Liquids, Wet Wastes, and Particulate Solids from the Nuclear Power Plant

Concentrated liquids, wet wastes, and particulate solids (Table 3.3.17) result from the various water cleanup operations conducted at nuclear power plants. These operations include ion exchange, filtration, and volume reduction by evaporation. The volume and total radioactivity estimated for these wastes are based on a recent survey.[10]

The resins used in demineralizers to remove ionic impurities from coolants are replaced periodically when their effectiveness declines. The service life of powdered resins is less than that of bead resins, which are used in deep beds and may be regenerated several times before they are replaced. Spent resins contain radioactive ions of fission product elements and activated corrosion products that have replaced ions of the resin matrix.[11]

Sludges are the thick mixtures (~40 wt% solids) of water and solid cake released from the blowback of filters used to remove suspended solids from reactor coolants. The cake may include cellulosic fibers or a powdered resin precoat, as well as solids collected from the coolant.

Filter cartridges are used to remove suspended solids from LWR coolants. Typical construction materials used in these filters include stainless steel frames and filter media of cloth, plastic or stainless steel fibers.

Liquid wastes from the nuclear power plant are usually decontaminated by evaporation,[12] which produces purified water that can be ultimately released to the environment and contaminated concentrates or slurries that must be solidified for shipment offsite. The main component of slurries from a BWR is Na_2SO_4 from resin regeneration; the main component of slurries from a PWR is $Na_2B_4O_7$, which is added to the reactor coolant as part of the reactivity control system.

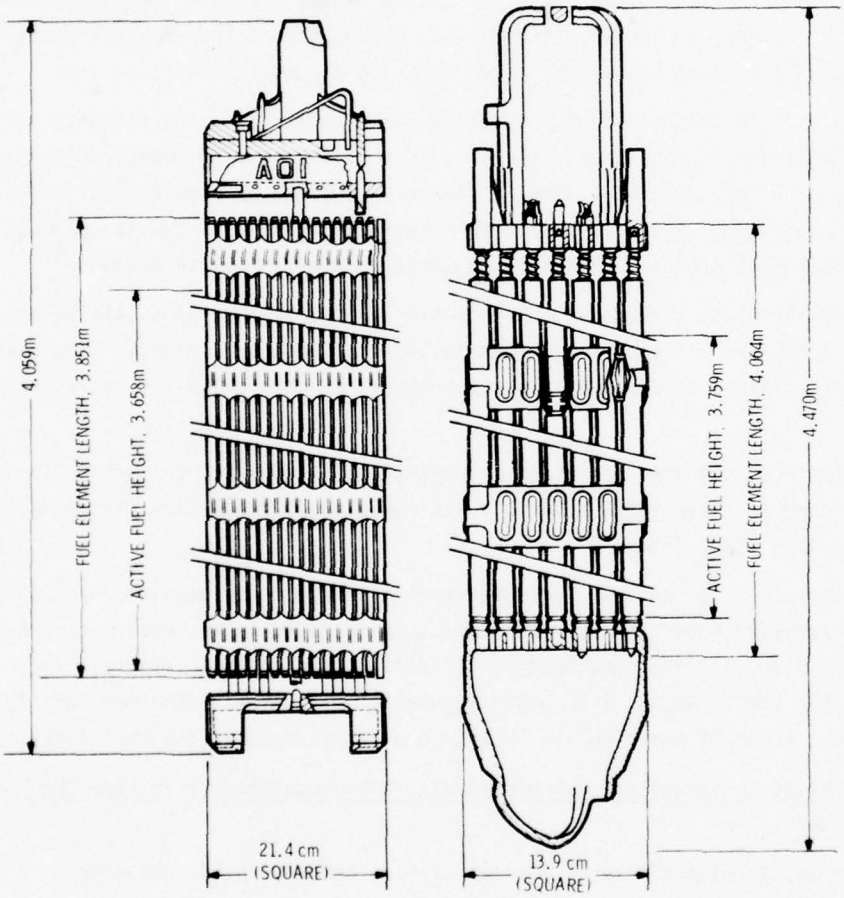
3.3.1.3 Failed Equipment and Noncompactable, Noncombustible Waste from the Nuclear Power Plant

The volume and total activity of noncompactable, noncombustible components of nuclear power plant waste (Table 3.3.18) are based on a previous characterization of trash.[8] Twenty percent of the total trash is assumed to be noncombustible. Failed plant equipment may include coolant reheaters, steam generators, turbines, piping, pumps, valves, seals, control rods, power-shaping rods, burnable poison rods, thimble plugs, flow channels, and in-core instrument

assemblies. The volume and total radioactivity of failed equipment is based on a recent estimate.[13] The amount of tritium in the failed equipment is based on estimates of the formation of tritium by neutron activation of boron-containing components[4] in a ratio of 2 PWR:1 BWR.

3.3.1.4 Spent Fuel

For the case in which LWR fuel is used on a once-through basis without recycle, the spent fuel (Table 3.3.22) is classed as a waste to be stored in surface facilities and ultimately placed in geologic repositories. Characterization of the spent fuel for this fuel cycle mode is based on descriptions by Kee, et al.[14] The reference PWR and BWR fuel assemblies are depicted in Figure 3.3.1.



	PWR	BWR
FUEL ELEMENT ARRAY	17 x 17	8 x 8
ASSEMBLY TOTAL WEIGHT, kg	658	283.9
URANIUM/ASSEMBLY, kg	461.4	188.7
MO_2/ASSEMBLY, kg	523.4	214.1
ZIRCALOY/ASSEMBLY, kg	108.4	59.6
HARDWARE/ASSEMBLY, kg	26.2	10.2
TOTAL METAL/ASSEMBLY, kg	134.6	69.8

FIGURE 3.3.1. Unirradiated Reference Fuel Assemblies

3.3.2 Independent Spent Fuel Storage Basin Primary Wastes

Data available for storage pools at the Nuclear Fuels Services fuel reprocessing plant;[15] General Electric's storage facility at Morris, Illinois;[16,17] the Allied General Nuclear Services fuel reprocessing plant;[18] and a proposed Exxon reprocessing facility[19] were used to characterize the waste streams (Tables 3.3.23 - 3.3.26) for the independent spent fuel storage basin (ISFSB). Waste streams at spent fuel storage basins derive from operations associated with the storage basin water treatment system, fuel cask unloading and loading operations, the building ventilation system, plant maintenance, the analytical laboratory, and general support operations. To estimate waste streams, the independent spent fuel storage basin was assumed to receive 500 MTHM/yr of spent fuel 0.5 years out of the reactor and to ship 500 MTHM/yr of spent fuel cooled 6.5 yr. A constant storage pool inventory of 3000 MTHM of spent fuel with a mean age of 3.5 yr was assumed.

3.3.2.1 Gaseous Waste from the Independent Spent Fuel Storage Basin

The three gaseous waste streams in the ISFSB are the heating, ventilation, and air conditioning (HVAC) stream; vaporized excess water; and process off-gas (Table 3.3.23). The HVAC stream basically comprises the ventilation air from the facility and has by far the largest volume. The only activity in this stream is a slight amount of tritium from leaking fuel elements. Excess water will be vaporized and dispersed to the atmosphere via the ISFSB stack. Process off-gas comprises the gaseous wastes from the cask handling area and from special vents used to collect the volatile gases from leaking fuel elements. These latter wastes are isolated so that special treatment can be applied to remove iodine.

3.3.2.2 Compactable and Combustible Wastes from the Independent Spent Fuel Storage Basin

The volume and radioactivity of the ISFSB combustible trash and ventilation filters (Table 3.3.24) were estimated from information in References 19 and 20, respectively.

3.3.2.3 Concentrated Liquids, Wet Wastes, and Particulate Solids from the Independent Spent Fuel Storage Basin

Cleanup or treatment of the basin water is responsible for most of the ISFSB wastes in the concentrated liquids, wet wastes, and particulate solids category (Table 3.3.25). The other source of waste in this category is the miscellaneous solution concentrates derived from shipping cask wastes, laboratory wastes and facility decontamination solutions.

3.3.2.4 Failed Equipment and Noncompactable, Noncombustible Waste from the Independent Spent Fuel Storage Basin

The volumes of ISFSB failed equipment and noncompactable, noncombustible waste (Table 3.3.26) are linearly scaled estimates from an Exxon preliminary safety analysis report.[19] No data on the radioactivity associated with these wastes are available, so the values given are best estimates.

3.3.3 Fuel Reprocessing Plant Primary Wastes

The reference fuel reprocessing plant (FRP) is a complex of four integrated facilities.* The waste streams (Tables 3.3.27 - 3.3.36) emanating from the FRP have the greatest variety of any wastes from the postfission LWR fuel cycle. The four integrated facilities include: 1) the spent fuel storage basin, 2) the fuel dissolution and solvent extraction facility, i.e. the main plant, 3) the plutonium conversion facility, and 4) the UF_6 conversion facility. Wastes generated in the spent fuel storage basin and parts of the UF_6 facility can be isolated and managed as nontransuranic wastes; the remainder of the FRP wastes are classified here as transuranic wastes.

3.3.3.1 Gaseous Wastes from the Fuel Reprocessing Plant

Six separate gaseous waste streams (Table 3.3.27) have been identified according to source for the FRP. Estimates of their characteristics were derived from pertinent literature as follows:

- fuel shear and dissolver off-gas - see Reference 22
- vessel off-gas - see Reference 22
- main plant HVAC - see Reference 23
- vaporized excess water - see Reference 8
- UF_6 plant process off-gas - see Reference 24
- storage basin HVAC - see Reference 8.

3.3.3.2 Fuel Residue

Fuel residue, comprising hulls and assembly hardware (Table 3.3.28), is unique to the FRP. The hardware is removed from the assemblies and separated prior to shearing the fuel. The radioactivity in this hardware is exclusively from neutron activation of its constituent metals; the radioactivity in the hulls includes activation products, as well as fission products and actinides from residual undissolved fuel.[25,26,27]

3.3.3.3 High-level Liquid Waste

The high-level liquid waste from the FRP (Table 3.3.29) comprises the concentrated raffinate from the first cycle extractor, combined with the dissolver sludge. This one stream contains over 99% of the fission products, as well as over 99% of the actinides (with the exception of uranium and plutonium) contained in the original fuel. One-half percent of the uranium and plutonium contained in the original fuel appear in the HLLW as waste losses.

3.3.3.4 Plutonium as Waste

Purified plutonium[28] becomes a waste (Table 3.3.30) in the uranium-only recycle option. The nitrate solution is used if the plutonium is to be solidified with HLLW; the oxide is used if the plutonium is to be stored as canisters of PuO_2.

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* Actually five facilities could be defined, since waste management requires sufficient plant space and equipment to be considered a separate facility. In the context of this report, however, wastes generated from waste management activities are not primary wastes but secondary wastes. Secondary wastes are discussed in Section 3.5.

3.3.3.5 Concentrated Liquids, Wet Wastes, and Particulate Solids from the Fuel Reprocessing Plant

Fuel reprocessing plant wastes in the concentrated liquids, wet wastes, and particulate solids category (Tables 3.3.31 and 3.3.32) occur as both TRU and non-TRU wastes. The wastes from the main plant and the fuel storage basin (Table 3.3.31) are generally of a significantly higher activity level than the low-level wastes from the UF_6 plant (Table 3.3.32).

The intermediate-level liquid waste (ILLW) stream (the major component of the wastes described in Table 3.3.31) consists of the concentrate produced in the general purpose concentrator and as such is made up of a catch-all mixture of aqueous wastes generated throughout the FRP. The ILLW is concentrated as an acidic solution; the ILLW shown in the table has been neutralized as required for some solidification processing. The major components of the ILLW are solvent cleanup wash solutions and solvent extraction battery raffinates, excluding the raffinate from the first solvent extraction contactor. Silica gel from the uranium purification process[23] is also included in the TRU class of wastes.

The non-TRU wastes in this category come from the FRP fuel storage basin. Their characteristics are very similar to the related wastes produced at the independent spent fuel storage basin. The bead resins are ion exchange resins used in the purification of storage basin water. The filter precoat sludge is the sludge removed from the basin water filters. The sulfate concentrate is derived from the resin regenerative solutions. The miscellaneous solution stream consists of decontamination washings and general laboratory wastes from the spent fuel storage basin.

The wastes originating in the UF_6 facility (Table 3.3.32) contain low levels of activity.[24] As stated above, these wastes can be divided into TRU and non-TRU fractions. The two streams making up the TRU wastes are bed residues and fines generated during operation of the fluid bed fluorinator. The non-TRU wastes result from off-gas cleanup. The K_2UO_4 mud is the insoluble residues remaining after KOH and KF are decanted from the cold trap off-gas cleanup solution. The decanted KOH-KF solution is combined with similar solutions from the hydrofluorinator off-gas scrubber and from the hydrogen vent scrubber in the fluorine plant. This composite solution is treated with lime to precipitate CaF_2 and recover KOH for recycle to the scrubbers. The resulting CaF_2 is filtered and dried to become UF_6 plant waste dryer discharge, the UF_6 waste stream with the largest volume.

3.3.3.6 Compactable and Combustible Wastes from the Fuel Reprocessing Plant

The FRP compactable and combustible wastes have been divided into two groups according to their gamma radiation levels. One group (Table 3.3.33) must be processed in a shielded facility designed for intermediate-level wastes; the other group (Table 3.3.34) can be processed in an unshielded facility for low-level wastes (LLW) only.

The volumes and radioactivities estimated for the waste streams are based on information contained in References 23, 29, and 30. The wastes are classified as TRU or non-TRU based on

source facility. The wastes from the fuel storage basin and UF_6 plant are non-TRU; the wastes from the main plant and PuO_2 conversion facility are TRU. All of the facilities generate combustible trash and ventilation filter wastes. The main plant also generates two unique waste streams included in the category of combustible wastes: 1) ion exchange bead resins and 2) degraded extractant.

The main plant ion exchange resins may be from the iodine removal process for the excess water or from solvent cleanup. While these ion exchange techniques are still under development, it is expected that one or both would be features of reprocessing plants; ion exchange resins, therefore, must be considered as constituents of the FRP wastes. The amount of FRP degraded extractant waste, however, may decrease to almost zero. Degraded extractant is that extractant which can no longer be used in the uranium-plutonium recovery process. Because of improved initial quality and cleanup techniques, the estimated annual volume of this waste (Table 3.3.33) is probably high.

The waste stream for unshielded processing is made up of a single stream of low-gamma, combustible trash originating in the main plant.

3.3.3.7 Failed Equipment and Noncompactable, Noncombustible Waste from the Fuel Reprocessing Plant

All of the FRP failed equipment and noncompactable, noncombustible waste from the main plant, the UF_6 plant, and the PuO_2 conversion facility (Table 3.3.35) is assumed to be TRU waste. The wastes in this category from the spent fuel storage basin (Table 3.3.36) are assumed to be non-TRU.

3.3.4 Mixed Oxide Fuel Fabrication Plant Primary Wastes

The principal activities in the mixed oxide fuel fabrication plant (MOX FFP) wastes (Tables 3.3.37 - 3.3.40) come from the plutonium that has been separated from spent fuel at the FRP and shipped to the fabrication plant. One year is assumed to elapse between reprocessing and refabrication of the fuel at the MOX FFP.

The feed material to the MOX FFP comprises dry oxides of natural uranium and plutonium. In the fabrication process line, these oxides are mixed, pelletized, and loaded into Zircaloy tubes to produce mixed oxide fuel elements, the final product of the plant. In the course of production, the four standard categories of wastes are generated. Each category is discussed separately below.

3.3.4.1 Gaseous Waste from the Mixed Oxide Fuel Fabrication Plant

The MOX FFP gaseous waste includes the plant air filtration stream and vaporized excess water. Most of the plant operations are conducted in isolated modules; the exhaust air from each module is filtered at least two times before being combined with building ventilation air to form the influent to the final air filtration system. The vaporized excess water bypasses the final air filtration system and is released directly to the stack.

3.3.4.2 Compactable and Combustible Wastes from the Mixed Oxide Fuel Fabrication Plant

MOX FFP compactable and combustible wastes (Table 3.3.38) from the glove box enclosures include packaging, bagging plastic, glove box gloves, and cleaning materials. Combustibles from the operating areas of the plant comprise much the same materials as found in the glove boxes but will also include surgical gloves, plastic sheeting, disposable protective clothing, and paper products. HEPA filters from the air filtration systems also fall into this waste category. Estimates of the actinide loading on the HEPA filters vary considerably.[31,32,33] The value given in Table 3.3.38 is a best judgment estimate.

3.3.4.3 Concentrated Liquids, Wet Wastes, and Particulate Solids from the Mixed Oxide Fuel Fabrication Plant

The volume of MOX FFP concentrated liquids and wet wastes (Table 3.3.39) is not large, as would be expected from an essentially dry process. Some process solution wastes are generated, however, mainly during fuel pellet washing, cladding etching, and decontamination.[34]

Scrap recovery generates the most MOX FFP aqueous waste. Approximately 2 wt% of the input UO_2-PuO_2 powder becomes so contaminated with impurities during the various steps of fuel fabrication that it cannot be recycled directly.[35] In the reference plant this dirty scrap is recovered using an aqueous recovery process. The principal waste stream from this process is the waste treatment evaporator bottoms.[36]

3.3.4.4 Failed Equipment and Noncompactable, Noncombustible Waste from the Mixed Oxide Fuel Fabrication Plant

The MOX FFP failed equipment and noncompactable, noncombustible wastes (Table 3.3.40) from the glove box enclosures are a mixture of scrap fuel cladding made of zirconium alloys, discarded equipment composed of austenitic steels and aluminum alloys, replaced enclosure ware (such as ventilation ducts and fittings of painted or galvanized steel and of aluminum), and metal packaging material and transfer containers (primarily of aluminum).

Noncombustible wastes from the restricted operating area outside the glove box enclosures are TRU-suspect and handled as TRU wastes. Examples include outer packaging cases, discarded tools and equipment, glass in the form of discarded laboratory ware, and empty reagent bottles and used heat-insulating material of fiber glass or mineral wool.[37]

REFERENCE RADIONUCLIDE INVENTORY TABLES

TABLE 3.3.4. Radionuclide Inventories in Liquid and Solid Wastes from the Reference Nuclear Power Plant and Spent Fuel Storage Basins, All Fuel Cycle Modes(a)

Activation Products	Nuclear Power Plant Wastes(b)		Independent Spent Fuel Storage Basin Wastes(c)				Fuel Reprocessing Plant Spent Fuel Storage Basin Wastes(d)	
	Ci/GWe-yr	Ci/yr	Receiving, Ci/MTHM(e)	Storing, Ci/MTHM-yr(f)	Shipping, Ci/MTHM(g)	Composite, Ci/yr(h)	Storing, Ci/MTHM-yr(i)	Composite, Ci/yr(j)
^{3}H	5.4×10^{2}	4.5×10^{2}						
^{51}Cr	4.5×10^{-1}	4.0×10^{-1}	2.2×10^{-4}	5.1×10^{-7}		1.1×10^{-1}	3.1×10^{-6}	4.6×10^{-1}
^{54}Mn	2.0×10^{1}	1.7×10^{1}	9.9×10^{-3}	2.5×10^{-4}	2.6×10^{-5}	5.7	8.5×10^{-4}	2.1×10^{1}
^{55}Fe	4.2×10^{2}	3.5×10^{2}	2.1×10^{-1}	1.3×10^{-2}	1.5×10^{-2}	1.5×10^{2}	2.3×10^{-2}	4.6×10^{2}
^{59}Fe	3.5	3.0	1.8×10^{-3}	6.5×10^{-6}	9.5×10^{-19}	9.0×10^{-1}	3.9×10^{-5}	3.6
^{58}Co	1.3×10^{2}	1.1×10^{2}	6.4×10^{-2}	3.8×10^{-4}	1.2×10^{-11}	3.3×10^{1}	2.2×10^{-3}	1.3×10^{2}
^{60}Co	2.6×10^{2}	2.2×10^{2}	1.3×10^{-1}	1.1×10^{-2}	2.0×10^{-2}	1.1×10^{2}	1.5×10^{-2}	2.9×10^{2}
Subtotal	1.4×10^{3}	1.1×10^{3}	4.2×10^{-1}	2.5×10^{-2}	3.5×10^{-2}	3.0×10^{2}	4.1×10^{-2}	9.1×10^{2}
Fission Products								
^{89}Sr	1.8×10^{1}	1.5×10^{1}	9.2×10^{-3}	3.8×10^{-5}	2.7×10^{-16}	4.7	2.3×10^{-4}	1.9×10^{1}
^{90}Sr	3.3×10^{1}	2.8×10^{1}	1.6×10^{-2}	1.9×10^{-3}	4.8×10^{-3}	1.6×10^{1}	2.0×10^{-3}	3.7×10^{1}
^{95}Zr	8.3×10^{-1}	6.9×10^{-1}	4.1×10^{-4}	2.2×10^{-6}	1.2×10^{-14}	2.1×10^{-1}	1.3×10^{-5}	8.5×10^{-1}
^{103}Ru	7.0×10^{-2}	6.0×10^{-2}	3.5×10^{-5}	1.1×10^{-7}	2.5×10^{-22}	1.8×10^{-2}	6.8×10^{-7}	7.1×10^{-2}
^{106}Ru	2.0	1.6	9.5×10^{-4}	2.8×10^{-5}	5.2×10^{-6}	5.6×10^{-1}	8.6×10^{-5}	2.1
^{127M}Te	8.2	6.9	4.1×10^{-3}	3.7×10^{-5}	1.2×10^{-9}	2.2	2.0×10^{-4}	8.6
^{129M}Te	9.0×10^{-1}	7.6×10^{-1}	4.5×10^{-4}	1.2×10^{-6}	2.7×10^{-24}	2.3×10^{-1}	7.4×10^{-6}	9.2×10^{1}
^{134}Cs	2.0×10^{3}	1.7×10^{3}	9.9×10^{-1}	5.3×10^{-2}	4.4×10^{-2}	6.8×10^{2}	1.0×10^{-1}	2.2×10^{3}
^{137}Cs	1.9×10^{3}	1.6×10^{3}	9.6×10^{-1}	1.1×10^{-1}	2.8×10^{-1}	9.5×10^{2}	1.2×10^{-1}	2.2×10^{3}
^{141}Ce	8.7×10^{-2}	7.0×10^{-2}	4.3×10^{-5}	1.2×10^{-7}		2.2×10^{2}	7.0×10^{-7}	8.8×10^{-2}
^{144}Ce	4.8	3.9	2.3×10^{-3}	5.4×10^{-5}	3.7×10^{-6}	1.3	1.9×10^{-4}	5.1
Subtotal	4.0×10^{3}	3.3×10^{3}	2.0	1.8×10^{-1}	3.3×10^{-1}	1.7×10^{3}	2.3×10^{-1}	4.4×10^{3}
Total Radionuclides	5.3×10^{3}	4.5×10^{3}	2.4	1.9×10^{-1}	3.6×10^{-1}	2.0×10^{3}	2.7×10^{-1}	5.3×10^{3}

a. Short-lived daughters are excluded.
b. The reference nuclear power plant operates at 1200 MWe for 70% of the time to produce 0.84 GWe-yr of electricity per year.
c. The independent spent fuel storage basin receives and ships 500 MT of fuel per year, with an inventory of 3000 MT.
d. The spent fuel storage basin attached to the fuel reprocessing plant receives, stores, and ships (to the FRP) 2000 MT/yr.
e. Based on estimated total inventory of 2.4 Ci/MTHM; 6 months cooling; same percent distribution as nuclear power plant.
f. Same as (e) except 6-month-cooled inventory = 0.3 Ci/MTHM-yr and activity is a weighted average during 6-yr storage.
g. Same as (e) except cooled 6.5 years and inventory = 0.8 Ci/MTHM, based on 6-month cooling.
h. Ci/yr = 500 (e+g) + 3000 (f).
i. Same as (f) except 1-yr storage.
j. Ci/yr = 2000 (e+i).

TABLE 3.3.5. Factors for Radioactive Decay of Activation and Fission Products in Liquid and Solid Wastes from the Reference Nuclear Power Plant and Spent Fuel Storage Basins[a]

Activation Products	Half-Life	Ci for Various Decay Periods							
		0	1 year	3 year	5 year	6 year	10^1 year	10^2 year	10^3 year
^{3}H	12.3 yr	1	9.45×10^{-1}	8.44×10^{-1}	7.5×10^{-1}	7.13×10^{-3}	5.69×10^{-1}	3.59×10^{-3}	
^{51}Cr	27.8 days	1	1.12×10^{-4}	1.40×10^{-12}					
^{54}Mn	312.5 days	1	4.45×10^{-1}	8.82×10^{-2}	1.75×10^{-2}	7.78×10^{-3}	3.06×10^{-4}		
^{55}Fe	2.7 yr	1	7.74×10^{-1}	4.63×10^{-1}	2.77×10^{-1}	2.14×10^{-1}	7.65×10^{-2}		
^{59}Fe	44.6 days	1	3.44×10^{-3}	4.08×10^{-8}					
^{58}Co	71.3 days	1	2.88×10^{-2}	2.39×10^{-5}	1.98×10^{-8}	5.70×10^{-10}			
^{60}Co	5.26 yr	1	8.77×10^{-1}	6.74×10^{-1}	5.18×10^{-1}	4.54×10^{-1}	2.68×10^{-1}	1.91×10^{-6}	
Fission Products									
^{89}Sr	52 days	1	7.72×10^{-3}	4.60×10^{-7}					
^{90}Sr	29 yr	1	9.76×10^{-1}	9.31×10^{-1}	8.87×10^{-1}	8.66×10^{-1}	7.87×10^{-1}	9.16×10^{-2}	4.17×10^{-11}
^{95}Zr	65.5 days	1	2.10×10^{-1}	9.30×10^{-6}	3.97×10^{-9}	8.65×10^{-11}			
^{103}Ru	39.6 days	1	1.68×10^{-3}	4.76×10^{-9}					
^{106}Ru	369 days	1	5.04×10^{-1}	1.28×10^{-1}	3.25×10^{-2}	1.64×10^{-2}	1.06×10^{-3}		
^{127m}Te	109 days	1	9.82×10^{-2}	9.47×10^{-4}	9.14×10^{-6}	8.98×10^{-7}			
^{129m}Te	33.6 days	1	5.38×10^{-4}	1.56×10^{-10}					
^{134}Cs	2.05 yr	1	7.13×10^{-1}	3.63×10^{-1}	1.84×10^{-1}	1.32×10^{-1}	3.40×10^{-2}		
^{137}Cs	30.0 yr	1	9.77×10^{-1}	9.33×10^{-1}	8.91×10^{-1}	8.71×10^{-1}	7.94×10^{-1}	9.93×10^{-2}	9.29×10^{-11}
^{141}Ce	32.4 days	1	4.07×10^{-4}	6.74×10^{-11}					
^{144}Ce	284.2 days	1	4.11×10^{-1}	6.92×10^{-2}	1.17×10^{-2}	4.80×10^{-3}	1.36×10^{-4}		

a. Multiply the curies of radionuclide initially in the waste by factors given in this table to obtain the curies of radionuclide remaining after the various decay periods.

TABLE 3.3.6. Activation Product Inventory in Reference Fuel Assembly Hardware,[a] as a Function of Decay Time, All Fuel Cycle Modes

Radionuclide	Ci/MTHM for Various Decay Periods[b]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{14}C	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	2.0×10^{-2}	1.0×10^{-7}	
^{35}S	8.0×10^{-2}	4.0×10^{-3}	1.0×10^{-5}	2.0×10^{-9}						
^{45}Ca	1.0×10^{-2}	3.0×10^{-3}	1.0×10^{-4}	1.0×10^{-6}	3.0×10^{-9}					
^{46}Sc	8.0×10^{-1}	4.0×10^{-2}	9.0×10^{-5}	1.0×10^{-8}						
^{54}Mn	3.0×10^{2}	2.0×10^{2}	3.0×10^{1}	2.0	8.0×10^{-2}					
^{55}Fe	6.0×10^{3}	4.0×10^{3}	2.5×10^{3}	1.0×10^{3}	4.0×10^{2}	1.0×10^{-8}				
^{58}Co	2.0×10^{3}	2.0	4.0×10^{-2}	1.0×10^{-6}						
^{60}Co	5.0×10^{3}	4.0×10^{3}	3.0×10^{3}	2.0×10^{3}	1.0×10^{3}	7.0×10^{-3}				
^{59}Ni	3.0	3.0	3.0	3.0	3.0	3.0	3.0	2.0	1.0	3.0×10^{-4}
^{63}Ni	4.0×10^{2}	4.0×10^{2}	4.0×10^{2}	4.0×10^{2}	4.0×10^{1}	2.0×10^{2}	2.0×10^{-1}			
^{65}Zn	2.0×10^{1}	6.0	8.0×10^{-5}	4.0×10^{-2}	6.0×10^{-4}					
^{90}Sr	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	9.0×10^{-4}	9.0×10^{-4}	9.0×10^{-5}				
^{90}Y	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	9.0×10^{-4}	9.0×10^{-4}	9.0×10^{-5}				
^{94}Nb	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	3.0×10^{-6}	
^{93}Mo	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	6.0×10^{-3}	1.0×10^{-3}		
^{99}Tc	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	7.0×10^{-3}	5.0×10^{-3}	2.6×10^{-4}
^{113m}Cd	4.0	4.0	3.0	2.0	2.0	3.0×10^{-2}				
^{125}Sb	4.0×10^{1}	2.0×10^{1}	2.0×10^{1}	8.0	2.0					
^{125m}Te	2.0×10^{1}	1.0×10^{1}	7.0	3.0	1.0					
^{181}W	4.0×10^{-4}	7.0×10^{-5}	2.0×10^{-6}	9.0×10^{-9}						
Total	1.3×10^{4}	8.7×10^{3}	5.5×10^{3}	3.4×10^{3}	1.8×10^{3}	2.0×10^{2}	3.3	2.0	1.0	5.6×10^{-4}
Total thermal watts	6.8×10^{1}	5.8×10^{1}	4.4×10^{1}	2.9×10^{1}	1.7×10^{1}	3.2×10^{-2}	1.2×10^{-4}	5.9×10^{-5}	4.0×10^{-5}	2.2×10^{-5}

a. Hardware components are 302-304 stainless steel, Inconel and Nicrobraz-50.
b. Periods are measured from reactor discharge.

TABLE 3.3.7. Activation Product Inventory in Reference Zircaloy Cladding Hulls as a Function of Decay Time, All Fuel Cycle Modes

Radionuclide	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{14}C	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	5.0×10^{-2}	2.0×10^{-2}	1.0×10^{-7}	
^{35}S	2.0×10^{-1}	9.0×10^{-3}	3.0×10^{-5}	6.0×10^{-9}						
^{54}Mn	4.0	2.0	3.0×10^{-1}	2.0×10^{-2}	8.0×10^{-4}					
^{55}Fe	1.0×10^{2}	9.0×10^{1}	5.0×10^{1}	2.0×10^{1}	8.0					
^{58}Co	6.0×10^{1}	2.0	2.0×10^{-3}	4.0×10^{-8}						
^{60}Co	2.0×10^{2}	1.0×10^{2}	1.0×10^{2}	8.0×10^{1}	4.0×10^{1}	3.0×10^{-4}				
^{59}Ni	3.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}	1.0×10^{-2}	3.0×10^{-6}
^{63}Ni	5.0	4.0	4.0	4.0	4.0	2.0	2.0×10^{-3}			
^{93}Zr	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	6.0×10^{-2}
^{95}Zr	4.0×10^{3}	8.0×10^{1}	3.0×10^{-2}	3.0×10^{-7}						
^{93m}Nb	9.0×10^{3}	1.0×10^{-2}	2.0×10^{-2}	3.0×10^{-2}	4.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	9.0×10^{-2}	6.0×10^{-2}
^{95}Nb	7.0×10^{3}	2.0×10^{2}	7.0×10^{-2}	6.0×10^{-7}						
^{113m}Cd	7.0×10^{-3}	5.0×10^{-3}	5.0×10^{-3}	3.0×10^{-3}	3.0×10^{-3}	4.0×10^{-5}				
^{119m}Sn	1.0×10^{1}	5.0	6.0×10^{-1}	3.0×10^{-2}	5.0×10^{-4}					
^{121m}Sn	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	1.0×10^{-1}	4.0×10^{-5}			
^{123}Sn	2.0×10^{-1}	2.0×10^{-2}	3.0×10^{-4}	8.0×10^{-7}						
^{181}W	1.0×10^{-3}	2.0×10^{-4}	5.0×10^{-6}	2.0×10^{-8}						
Total	1.1×10^{4}	4.8×10^{2}	1.6×10^{2}	1.0×10^{2}	5.3×10^{1}	3.7×10^{-1}	2.6×10^{-1}	2.3×10^{-1}	1.9×10^{-1}	1.2×10^{-1}
Total thermal watts	3.9	2.3	1.8	1.2	3.3×10^{-1}	3.2×10^{-4}	2.5×10^{-5}	1.6×10^{-5}	1.0×10^{-5}	6.9×10^{-6}

a. Periods are measured from reactor discharge.

TABLE 3.3.8. Fission Products in Reference Spent Fuel as a Function of Decay Time, Once-Through and Uranium-Only Recycle (Case 1 and 2)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{3}H	4.4×10^{2}	4.2×10^{2}	3.7×10^{2}	3.1×10^{2}	2.5×10^{2}	1.6				
^{14}C(b,c)	7.4×10^{-1}	7.4×10^{-1}	7.4×10^{-1}	7.4×10^{-1}	7.4×10^{-1}	7.3×10^{-1}	6.5×10^{-1}	2.2×10^{-1}	4.2×10^{-6}	
^{14}C(b,d)	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	5.5×10^{-1}	1.9×10^{-1}	3.5×10^{-6}	
^{79}Se	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.2×10^{-1}	1.2×10^{-1}	8.2×10^{-6}
^{85}Kr	9.5×10^{3}	8.9×10^{3}	7.9×10^{3}	6.5×10^{3}	5.0×10^{3}	1.6×10^{1}				
^{87}Rb	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}
^{89}Sr	5.5×10^{4}	4.2×10^{2}	2.5×10^{-2}	1.1×10^{-8}						
^{90}Sr	6.7×10^{4}	6.5×10^{4}	6.3×10^{4}	5.8×10^{4}	5.2×10^{4}	5.7×10^{3}	1.3×10^{-6}			
^{90}Y	6.7×10^{4}	6.5×10^{4}	6.3×10^{4}	5.8×10^{4}	5.2×10^{4}	5.7×10^{3}	1.3×10^{-6}			
^{91}Y	9.5×10^{4}	1.3×10^{3}	2.4×10^{-1}	5.8×10^{-7}						
^{93}Zr	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.6	1.1
^{95}Zr	1.7×10^{5}	3.4×10^{3}	1.4	1.2×10^{-5}						
^{93m}Nb	1.7×10^{-1}	2.5×10^{-1}	3.9×10^{-1}	5.7×10^{-1}	7.7×10^{-1}	1.7	1.7	1.7	1.6	1.1
^{95m}Nb	3.6×10^{3}	7.3×10^{1}	3.0×10^{-2}	2.6×10^{-7}						
^{95}Nb	3.3×10^{5}	7.6×10^{3}	3.2	2.7×10^{-5}						
^{99}Tc	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.2×10^{1}	9.2	4.9×10^{-1}
^{103}Ru	4.3×10^{4}	7.2×10^{1}	2.0×10^{-4}	9.5×10^{-13}						
^{106}Ru	3.4×10^{5}	1.7×10^{5}	4.2×10^{4}	5.3×10^{3}	3.4×10^{2}					
^{103m}Rh	4.3×10^{4}	7.2×10^{1}	2.0×10^{-4}	9.5×10^{-13}						
^{106}Rh	3.4×10^{5}	1.7×10^{5}	4.2×10^{4}	5.3×10^{3}	3.4×10^{2}					
^{107}Pd	9.2×10^{-2}	9.9×10^{-2}	9.8×10^{-2}	9.8×10^{-2}	9.9×10^{-2}	9.9×10^{-2}	9.9×10^{-2}	9.9×10^{-2}	9.8×10^{-2}	9.0×10^{-2}
^{110m}Ag	1.8×10^{3}	6.6×10^{2}	8.8×10^{1}	4.4	8.1×10^{-2}					
^{110}Ag	2.3×10^{2}	8.6×10^{1}	1.1×10^{1}	5.7×10^{-1}	1.0×10^{-2}					
^{113m}Cd	1.2×10^{1}	1.1×10^{1}	7.2	6.2	7.0	8.2×10^{-2}				
^{119m}Sn	8.6	3.1	4.1×10^{-1}	2.0×10^{-2}	3.4×10^{-4}					
^{121m}Sn	4.6×10^{-4}	4.6×10^{-4}	4.4×10^{-4}	4.3×10^{-4}	4.2×10^{-4}	1.9×10^{-4}	5.1×10^{-8}			
^{123}Sn	2.7×10^{3}	3.6×10^{2}	6.3	1.4×10^{-2}	4.4×10^{-6}					
^{126}Sn	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.5×10^{-1}	2.4×10^{-1}	4.7×10^{-4}

a. Periods are measured from reactor discharge.
b. Not a fission product, ^{14}C is formed by neutron activation of ^{14}N impurity in fuel.
c. Once-through.
d. Uranium-only recyle.

TABLE 3.3.8. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{124}Sb	4.3×10^{1}	6.4×10^{-1}	1.4×10^{-4}	4.4×10^{-10}						
^{125}Sb	6.8×10^{3}	5.3×10^{3}	3.1×10^{3}	1.4×10^{3}	5.3×10^{2}	4.9×10^{-8}				
^{126m}Sb	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.5×10^{-1}	2.4×10^{-1}	4.7×10^{-4}
^{126}Sb	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.5×10^{-1}	2.4×10^{-1}	4.7×10^{-4}
^{123m}Te	1.6×10^{-1}	1.8×10^{-2}	2.3×10^{-4}	3.5×10^{-7}						
^{125m}Te	2.8×10^{3}	2.2×10^{3}	1.3×10^{3}	6.0×10^{2}	2.2×10^{2}	2.0×10^{-8}				
^{127m}Te	4.2×10^{3}	4.2×10^{2}	4.0	3.8×10^{-3}	3.5×10^{-7}					
^{127}Te	4.2×10^{3}	4.1×10^{2}	3.9	3.7×10^{-3}	3.5×10^{-7}					
^{129}I	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
^{134}Cs	1.7×10^{5}	1.2×10^{5}	6.0×10^{4}	2.2×10^{4}	5.7×10^{3}					
^{135}Cs	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.6×10^{-1}	2.1×10^{-1}
^{137}Cs	9.4×10^{4}	9.2×10^{4}	8.8×10^{4}	8.2×10^{4}	7.5×10^{4}	9.4×10^{3}	8.8×10^{-6}			
^{137m}Ba	8.8×10^{4}	8.6×10^{4}	8.2×10^{4}	7.8×10^{4}	7.0×10^{4}	8.8×10^{3}	8.2×10^{-6}			
^{141}Ce	2.4×10^{4}	9.7	1.6×10^{-6}	1.1×10^{-16}						
^{144}Ce	6.1×10^{5}	2.5×10^{5}	4.2×10^{4}	2.9×10^{3}	8.2×10^{1}					
^{144}Pr	6.1×10^{5}	2.4×10^{5}	4.2×10^{4}	2.9×10^{3}	8.2×10^{1}					
^{147}Pm	9.0×10^{4}	6.9×10^{4}	4.1×10^{4}	1.9×10^{4}	6.4×10^{3}	2.9×10^{-7}				
^{148m}Pm	1.7×10^{3}	4.1	2.4×10^{-5}	3.4×10^{-13}						
^{148}Pm	1.4×10^{2}	3.3×10^{-1}	1.9×10^{-6}	2.7×10^{-14}						
^{151}Sm	1.2×10^{3}	1.1×10^{3}	1.1×10^{3}	1.1×10^{3}	1.1×10^{3}	5.2×10^{2}	4.0×10^{-1}			
^{152}Eu	1.3×10^{1}	1.2×10^{1}	1.1×10^{1}	9.0	7.1	3.9×10^{-2}				
^{154}Eu	5.8×10^{3}	5.5×10^{3}	5.0×10^{3}	4.4×10^{3}	3.7×10^{3}	7.6×10^{1}				
^{155}Eu	5.7×10^{3}	3.9×10^{3}	1.8×10^{3}	5.7×10^{2}	1.2×10^{2}					
^{153}Gd	1.9×10^{1}	6.6	8.8×10^{-1}	3.5×10^{-2}						
^{160}Tb	1.8×10^{2}	5.2	4.6×10^{-3}	1.2×10^{-7}						
^{166m}Ho	5.4×10^{-4}	5.4×10^{-4}	5.2×10^{-4}	5.2×10^{-4}	5.4×10^{-4}	5.1×10^{-4}	3.0×10^{-4}	1.7×10^{-6}		
Total	3.3×10^{6}	1.4×10^{6}	5.9×10^{5}	3.5×10^{5}	2.7×10^{5}	3.0×10^{4}	1.9×10^{1}	1.8×10^{1}	1.4×10^{1}	2.9
Total Thermal Watts	1.5×10^{4}	6.3×10^{3}	2.5×10^{3}	1.2×10^{3}	8.9×10^{2}	9.2×10^{1}	2.0×10^{-2}	1.9×10^{-2}	1.2×10^{-2}	7.8×10^{-4}

a. Periods are measured from reactor discharge.

TABLE 3.3.9. Fission Products in Reference Spent Fuel as a Function of Decay Time, Uranium and Plutonium Recycle (Case 3)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{3}H	4.5×10^{2}	4.2×10^{2}	3.8×10^{2}	3.2×10^{2}	2.6×10^{2}	1.6				
^{14}C(b)	5.6×10^{-1}	5.6×10^{-1}	5.6×10^{-1}	5.6×10^{-1}	5.6×10^{-1}	5.6×10^{-1}	5.0×10^{-1}	1.7×10^{-1}	3.2×10^{-6}	
^{79}Se	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.1×10^{-1}	1.2×10^{-1}	8.0×10^{-6}
^{85}Kr	8.9×10^{3}	8.4×10^{3}	7.4×10^{3}	6.1×10^{3}	4.7×10^{3}	1.5×10^{1}				
^{87}Rb	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}
^{89}Sr	5.3×10^{4}	4.1×10^{2}	2.4×10^{-2}	1.1×10^{-8}						
^{90}Sr	6.2×10^{4}	6.1×10^{4}	5.8×10^{4}	5.8×10^{4}	4.9×10^{4}	5.3×10^{3}	1.2×10^{-6}			
^{90}Y	6.2×10^{4}	6.1×10^{4}	5.8×10^{4}	5.4×10^{4}	4.9×10^{4}	5.3×10^{3}	1.2×10^{-6}			
^{91}Y	9.3×10^{4}	1.3×10^{3}	2.3×10^{-1}	5.6×10^{-7}						
^{93}Zr	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.5	1.0
^{95}Zr	1.7×10^{5}	3.4×10^{3}	1.4	1.2×10^{-5}						
^{93m}Nb	1.7×10^{-1}	2.4×10^{-1}	3.7×10^{-1}	5.4×10^{-1}	7.4×10^{-1}	1.6	1.6	1.6	1.5	1.0
^{95m}Nb	3.6×10^{3}	7.3×10^{1}	3.0×10^{-2}	2.6×10^{-7}						
^{95}Nb	3.3×10^{5}	7.6×10^{3}	3.2	2.7×10^{-5}						
^{99}Tc	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.2×10^{1}	9.3	4.9×10^{-1}
^{103}Ru	4.4×10^{4}	7.4×10^{1}	2.1×10^{-4}							
^{106}Ru	3.7×10^{5}	1.9×10^{5}	4.7×10^{4}	5.9×10^{3}	3.8×10^{2}					
^{103m}Rh	4.4×10^{4}	7.4×10^{1}	2.1×10^{-4}							
^{106}Rh	3.7×10^{5}	1.9×10^{5}	4.7×10^{4}	5.9×10^{3}	3.8×10^{2}					
^{107}Pd	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.2×10^{-1}	1.1×10^{-1}
^{110m}Ag	2.2×10^{3}	8.2×10^{2}	1.1×10^{2}	5.5	1.0×10^{-1}					
^{110}Ag	2.9×10^{2}	1.1×10^{2}	1.4×10^{1}	7.1×10^{-1}	1.3×10^{-2}					
^{113m}Cd	1.1×10^{1}	1.0×10^{1}	9.4	8.1	6.6	7.7×10^{-2}				
^{119m}Sn	9.1	3.3	4.3×10^{-1}	2.1×10^{-2}	3.6×10^{-4}					
^{121m}Sn	5.0×10^{-4}	5.0×10^{-4}	4.9×10^{-4}	4.8×10^{-4}	4.6×10^{-4}	2.0×10^{-4}	5.5×10^{-8}			
^{123}Sn	2.9×10^{3}	3.8×10^{2}	6.6	1.5×10^{-2}	4.6×10^{-6}					
^{126}Sn	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.4×10^{-1}	5.1×10^{-1}	2.7×10^{-1}	5.4×10^{-4}
^{124}Sb	4.8×10^{1}	7.0×10^{-1}	1.5×10^{-4}							
^{125}Sb	7.6×10^{3}	5.9×10^{3}	3.5×10^{3}	1.6×10^{3}	5.8×10^{2}	5.4×10^{-8}				

a. Periods measured from reactor discharge.
b. Not a fission product, ^{14}C is formed by neutron activation of ^{14}N impurity in fuel.

TABLE 3.3.9. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{126m}Sb	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.5×10^{-1}	5.4×10^{-1}	5.1×10^{-1}	2.7×10^{-1}	5.4×10^{-4}
^{126}Sb	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.1×10^{-1}	2.7×10^{-1}	5.3×10^{-4}
^{123m}Te	1.5×10^{-1}	1.7×10^{-2}	2.2×10^{-4}	3.4×10^{-7}						
^{125m}Te	3.1×10^{3}	2.4×10^{3}	1.5×10^{3}	6.8×10^{2}	2.4×10^{2}	2.3×10^{-8}				
^{127m}Te	4.4×10^{3}	4.3×10^{2}	4.2	3.9×10^{-3}	3.6×10^{-7}					
^{127}Te	4.4×10^{3}	4.3×10^{2}	4.1	3.9×10^{-3}	3.6×10^{-7}					
^{129}I	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.5×10^{-2}	3.4×10^{-2}
^{134}Cs	1.7×10^{5}	1.2×10^{5}	6.1×10^{4}	2.2×10^{4}	5.7×10^{3}					
^{135}Cs	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	2.5×10^{-1}
^{137}Cs	9.5×10^{4}	9.3×10^{4}	8.9×10^{4}	8.3×10^{4}	7.5×10^{4}	9.4×10^{3}	8.8×10^{-6}			
^{137m}Ba	8.9×10^{4}	8.7×10^{4}	8.3×10^{4}	7.7×10^{4}	7.1×10^{4}	8.8×10^{3}	8.3×10^{-6}			
^{141}Ce	2.4×10^{4}	9.8	1.6×10^{-6}							
^{144}Ce	5.9×10^{5}	2.4×10^{5}	4.1×10^{4}	2.8×10^{3}	8.0×10^{1}					
^{144}Pr	5.9×10^{5}	2.4×10^{5}	4.1×10^{4}	2.8×10^{3}	8.0×10^{1}					
^{147}Pm	9.0×10^{4}	6.9×10^{4}	4.1×10^{4}	1.8×10^{4}	6.4×10^{3}	2.9×10^{-7}				
^{148m}Pm	1.8×10^{3}	4.4	2.6×10^{-5}							
^{148}Pm	1.5×10^{2}	3.5×10^{-1}	2.1×10^{-6}							
^{151}Sm	1.2×10^{3}	1.2×10^{3}	1.2×10^{3}	1.2×10^{3}	1.1×10^{3}	5.5×10^{2}	4.2×10^{-1}			
^{152}Eu	1.7×10^{1}	1.6×10^{1}	1.4×10^{1}	1.2×10^{1}	9.3	5.1×10^{-2}				
^{154}Eu	6.4×10^{3}	6.1×10^{3}	5.6×10^{3}	4.9×10^{3}	4.1×10^{3}	8.4×10^{1}				
^{155}Eu	6.2×10^{3}	4.2×10^{3}	2.0×10^{3}	6.2×10^{2}	1.3×10^{2}					
^{153}Gd	1.7×10^{1}	6.0	7.3×10^{-1}	3.2×10^{-2}	4.9×10^{-4}					
^{160}Tb	2.1×10^{2}	6.2	5.5×10^{-3}	1.5×10^{-7}						
^{166m}Ho	6.7×10^{-4}	6.7×10^{-4}	6.7×10^{-4}	6.7×10^{-4}	6.7×10^{-4}	6.3×10^{-4}	3.8×10^{-4}	2.1×10^{-6}		
Total	3.3×10^{6}	1.4×10^{6}	5.9×10^{5}	3.4×10^{5}	2.7×10^{5}	3.0×10^{4}	1.9×10^{1}	1.8×10^{1}	1.4×10^{1}	2.9
Total Thermal Watts	1.5×10^{4}	6.5×10^{3}	2.5×10^{3}	1.2×10^{3}	8.7×10^{2}	9.0×10^{1}	2.1×10^{-2}	2.0×10^{-2}	1.3×10^{-2}	7.9×10^{-4}

a. Periods are measured from reactor discharge.

TABLE 3.3.10. Actinides and Daughters in Reference Spent Fuel as a Function of Decay Time, Once-Through Cycle (Case 1)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb(b)					7.0×10^{-10}	1.7×10^{-6}	1.0×10^{-3}	5.0×10^{-2}	4.3×10^{-1}	3.6×10^{-1}
^{226}Ra(c)					7.4×10^{-9}	3.4×10^{-6}	1.0×10^{-3}	5.0×10^{-2}	4.3×10^{-1}	3.6×10^{-1}
^{227}Ac(d)					1.1×10^{-6}	2.6×10^{-5}	3.5×10^{-4}	3.4×10^{-3}	2.2×10^{-2}	2.6×10^{-2}
^{228}Th(e)	1.1×10^{-3}	2.5×10^{-3}	5.7×10^{-3}	9.7×10^{-3}	1.3×10^{-2}	6.4×10^{-3}	1.1×10^{-6}	1.3×10^{-7}	1.7×10^{-6}	1.7×10^{-5}
^{229}Th(f)					2.8×10^{-8}	9.0×10^{-7}	1.2×10^{-4}	1.5×10^{-2}	3.5×10^{-1}	8.6×10^{-1}
^{230}Th					4.1×10^{-6}	2.2×10^{-4}	6.0×10^{-3}	6.5×10^{-2}	4.2×10^{-2}	3.6×10^{-1}
^{231}Th	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.7×10^{-2}	1.9×10^{-2}	2.6×10^{-2}	2.7×10^{-2}
^{232}Th(g)					1.1×10^{-10}	1.1×10^{-9}	1.1×10^{-8}	1.3×10^{-7}	1.7×10^{-6}	1.7×10^{-5}
^{234}Th	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{231}Pa	1.8×10^{-6}	2.1×10^{-6}	2.8×10^{-6}	3.9×10^{-6}	5.3×10^{-6}	3.7×10^{-5}	3.5×10^{-4}	3.4×10^{-3}	2.2×10^{-2}	2.7×10^{-2}
^{233}Pa	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	4.1×10^{-1}	9.5×10^{-1}	1.1	1.1	8.1×10^{-1}
^{234m}Pa	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{234}Pa	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}
^{232}U	4.2×10^{-3}	6.7×10^{-3}	1.0×10^{-2}	1.3×10^{-2}	1.4×10^{-2}	6.2×10^{-3}	1.1×10^{-6}			
^{233}U	2.3×10^{-5}	2.5×10^{-5}	2.8×10^{-5}	3.2×10^{-5}	3.8×10^{-5}	1.8×10^{-4}	3.0×10^{-3}	4.5×10^{-2}	3.8×10^{-1}	8.6×10^{-1}
^{234}U	1.5×10^{-2}	2.1×10^{-2}	3.3×10^{-2}	5.1×10^{-2}	7.4×10^{-2}	4.4×10^{-1}	8.0×10^{-1}	7.9×10^{-1}	6.8×10^{-1}	3.4×10^{-1}
^{235}U	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.7×10^{-2}	1.9×10^{-2}	2.6×10^{-2}	2.7×10^{-2}

a. Periods are measured from reactor discharge.

b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.10. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}	2.4×10^{-1}	3.1×10^{-1}	3.5×10^{-1}	3.4×10^{-1}
^{237}U	2.8	2.7	2.4	2.1	1.7	2.6×10^{-2}	4.1×10^{-6}	1.9×10^{-6}	1.0×10^{-9}	
^{238}U	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{237}Np	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	3.1×10^{-1}	4.1×10^{-1}	9.5×10^{-1}	1.1	1.1	8.1×10^{-1}
^{239}Np	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.3×10^{1}	5.7	1.6×10^{-3}	1.2×10^{-7}
^{236}Pu	2.9×10^{-1}	2.3×10^{-1}	1.4×10^{-1}	6.8×10^{-2}	2.6×10^{-2}					
^{238}Pu	2.1×10^{3}	2.1×10^{3}	2.1×10^{3}	2.1×10^{3}	2.0×10^{3}	1.0×10^{3}	1.1			
^{239}Pu	2.9×10^{2}	2.9×10^{2}	2.9×10^{2}	2.9×10^{2}	2.9×10^{2}	2.9×10^{2}	2.8×10^{2}	2.2×10^{2}	1.7×10^{1}	1.2×10^{-7}
^{240}Pu	4.5×10^{2}	4.5×10^{2}	4.5×10^{2}	4.5×10^{2}	4.5×10^{2}	4.5×10^{2}	4.1×10^{2}	1.6×10^{2}	1.6×10^{-2}	1.3×10^{-9}
^{241}Pu	1.1×10^{5}	1.1×10^{5}	9.2×10^{4}	8.4×10^{4}	6.9×10^{4}	1.0×10^{3}	1.7×10^{-1}	7.8×10^{-2}	4.1×10^{-5}	
^{242}Pu	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.4	2.6×10^{-1}
^{241}Am	2.0×10^{2}	3.7×10^{2}	6.9×10^{2}	1.1×10^{3}	1.6×10^{3}	3.5×10^{3}	8.3×10^{2}	7.8×10^{-2}	4.1×10^{-5}	
^{242m}Am	1.1×10^{1}	1.1×10^{1}	1.1×10^{1}	1.0×10^{1}	1.0×10^{1}	6.9	1.1×10^{-1}			
^{242}Am	1.1×10^{1}	1.1×10^{1}	1.1×10^{1}	1.0×10^{1}	1.0×10^{1}	6.9	1.1×10^{-1}			
^{243}Am	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.3×10^{1}	5.7	1.6×10^{-3}	1.2×10^{-7}
^{242}Cm	1.7×10^{4}	3.6×10^{3}	1.7×10^{2}	1.0×10^{1}	8.5	5.6	9.3×10^{-2}			
^{243}Cm	4.0	3.9	3.8	3.5	3.2	4.6×10^{-1}	1.6×10^{-9}			
^{244}Cm	1.3×10^{3}	1.3×10^{3}	1.2×10^{3}	1.0×10^{3}	9.0×10^{2}	2.9×10^{1}				
^{245}Cm	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.7×10^{-1}	7.8×10^{-2}	4.1×10^{-5}	
^{246}Cm	3.5×10^{-2}	3.5×10^{-2}	3.4×10^{-2}	3.4×10^{-2}	3.5×10^{-2}	3.4×10^{-2}	3.0×10^{-2}	7.9×10^{-3}	1.4×10^{-8}	
^{249}Bk	1.1×10^{-3}	5.0×10^{-4}	9.9×10^{-5}	8.9×10^{-6}	3.5×10^{-7}					
^{249}Cf	1.9×10^{-6}	3.4×10^{-6}	4.4×10^{-6}	4.6×10^{-6}	4.6×10^{-6}	3.8×10^{-6}	6.5×10^{-7}			
^{250}Cf	1.7×10^{-5}	1.6×10^{-5}	1.4×10^{-5}	1.2×10^{-5}	9.9×10^{-6}	8.4×10^{-8}				
^{252}Cf	2.0×10^{-5}	1.5×10^{-5}	9.0×10^{-6}	4.1×10^{-6}	1.4×10^{-6}					
Total	1.3×10^{5}	1.1×10^{5}	1.0×10^{5}	8.9×10^{4}	7.5×10^{4}	6.3×10^{3}	1.6×10^{3}	4.0×10^{2}	3.1×10^{1}	1.5×10^{1}
Total Thermal Watts	7.7×10^{2}	2.9×10^{2}	1.7×10^{2}	1.7×10^{2}	1.8×10^{2}	1.7×10^{2}	5.0×10^{1}	1.2×10^{1}	8.3×10^{-1}	3.5×10^{-1}

a. Periods are measured from reactor discharge.

TABLE 3.3.11. Actinides and Daughters in Reference Spent Fuel as a Function of Decay Time, Uranium-Only Recycle (Case 2)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb(b)					1.21×10^{-9}	2.57×10^{-6}	1.52×10^{-3}	7.44×10^{-2}	6.17×10^{-1}	3.93×10^{-1}
^{226}Ra(c)					1.24×10^{-8}	5.12×10^{-6}	1.52×10^{-3}	7.44×10^{-2}	6.17×10^{-1}	3.93×10^{-1}
^{227}Ac(d)					1.19×10^{-6}	2.73×10^{-5}	3.63×10^{-4}	3.52×10^{-3}	2.21×10^{-2}	2.71×10^{-2}
^{228}Th(e)	1.83×10^{-3}	3.99×10^{-3}	9.05×10^{-3}	1.53×10^{-2}	1.98×10^{-2}	1.01×10^{-2}	1.76×10^{-6}	1.81×10^{-7}	2.14×10^{-6}	2.17×10^{-5}
^{229}Th(f)					3.36×10^{-8}	1.23×10^{-6}	1.50×10^{-4}	1.77×10^{-2}	3.95×10^{-1}	9.80×10^{-1}
^{230}Th	4.31×10^{-7}	7.01×10^{-7}	1.47×10^{-6}	3.19×10^{-6}	6.51×10^{-6}	3.27×10^{-4}	8.88×10^{-3}	9.56×10^{-2}	6.10×10^{-1}	3.92×10^{-1}
^{231}Th	1.69×10^{-2}	1.69×10^{-2}	1.69×10^{-2}	1.69×10^{-2}	1.69×10^{-2}	1.70×10^{-2}	1.72×10^{-2}	1.94×10^{-2}	2.65×10^{-2}	2.71×10^{-2}
^{232}Th(g)					1.57×10^{-10}	1.57×10^{-9}	1.60×10^{-8}	1.81×10^{-7}	2.14×10^{-6}	2.17×10^{-5}
^{234}Th	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.16×10^{-1}
^{231}Pa	2.01×10^{-6}	2.37×10^{-6}	3.09×10^{-6}	4.18×10^{-6}	5.63×10^{-6}	3.81×10^{-5}	3.63×10^{-4}	3.51×10^{-3}	2.21×10^{-2}	2.71×10^{-2}
^{233}Pa	4.67×10^{-1}	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.70×10^{-1}	5.62×10^{-1}	1.10	1.26	1.23	9.16×10^{-1}
^{234m}Pa	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.16×10^{-1}
^{234}Pa	3.15×10^{-4}	3.15×10^{-4}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-4}	3.15×10^{-4}	3.15×10^{-4}	3.15×10^{-4}	3.15×10^{-4}	3.16×10^{-4}
^{232}U	6.80×10^{-3}	1.06×10^{-2}	1.59×10^{-2}	1.99×10^{-2}	2.17×10^{-2}	9.83×10^{-3}	1.70×10^{-6}	4.08×10^{-44}		
^{233}U	2.53×10^{-5}	2.75×10^{-5}	3.19×10^{-5}	3.86×10^{-5}	4.75×10^{-5}	2.45×10^{-4}	3.68×10^{-3}	5.08×10^{-2}	4.32×10^{-1}	9.72×10^{-1}
^{234}U	2.69×10^{-2}	3.56×10^{-2}	5.32×10^{-2}	7.91×10^{-2}	1.13×10^{-1}	6.48×10^{-1}	1.18	1.16	9.68×10^{-1}	3.68×10^{-1}
^{235}U	1.69×10^{-2}	1.69×10^{-2}	1.69×10^{-2}	1.69×10^{-2}	1.69×10^{-2}	1.70×10^{-2}	1.72×10^{-2}	1.94×10^{-2}	2.65×10^{-2}	2.71×10^{-2}

a. Periods are measured from reactor discharge.
b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.
c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.
d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.
e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.
f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.
g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.11. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U	3.20×10^{-1}	3.20×10^{-1}	3.20×10^{-1}	3.20×10^{-1}	3.20×10^{-1}	3.21×10^{-1}	3.32×10^{-1}	4.02×10^{-1}	4.47×10^{-1}	4.36×10^{-1}
^{237}U	2.76	2.62	2.39	2.08	1.72	2.55×10^{-2}	3.95×10^{-6}	1.86×10^{-6}	9.80×10^{-10}	1.65×10^{-42}
^{238}U	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.15×10^{-1}	3.16×10^{-1}
^{237}Np	4.67×10^{-1}	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.70×10^{-1}	5.62×10^{-1}	1.10	1.26	1.23	9.16×10^{-1}
^{239}Np	1.36×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{236}Pu	4.60×10^{-1}	3.61×10^{-1}	2.22×10^{-1}	1.07×10^{-1}	4.04×10^{-2}	1.27×10^{-11}				
^{238}Pu	3.10×10^{3}	3.14×10^{3}	3.11×10^{3}	3.04×10^{3}	2.94×10^{3}	1.46×10^{3}	1.53	3.37×10^{-19}		
^{239}Pu	2.92×10^{2}	2.92×10^{2}	2.92×10^{2}	2.92×10^{2}	2.92×10^{2}	2.91×10^{2}	2.84×10^{2}	2.22×10^{2}	1.74×10^{1}	1.11×10^{-7}
^{240}Pu	4.52×10^{2}	4.52×10^{2}	4.52×10^{2}	4.52×10^{2}	4.53×10^{2}	4.51×10^{2}	4.11×10^{2}	1.63×10^{2}	1.61×10^{-2}	1.17×10^{-9}
^{241}Pu	1.10×10^{5}	1.05×10^{5}	9.54×10^{4}	8.29×10^{4}	6.88×10^{4}	1.02×10^{3}	1.58×10^{-1}	7.42×10^{-2}	3.92×10^{-5}	6.62×10^{-38}
^{242}Pu	1.58	1.58	1.58	1.58	1.58	1.58	1.58	1.55	1.32	2.54×10^{-1}
^{241}Am	1.94×10^{2}	3.66×10^{2}	6.84×10^{2}	1.11×10^{3}	1.58×10^{3}	3.44×10^{3}	8.24×10^{2}	7.47×10^{-2}	3.92×10^{-5}	6.98×10^{-38}
^{242m}Am	1.07×10^{1}	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.79	1.12×10^{-1}	1.70×10^{-19}		
^{242}Am	1.07×10^{1}	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.79	1.12×10^{-1}	1.70×10^{-19}		
^{243}Am	1.36×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{242}Cm	1.64×10^{4}	3.48×10^{3}	1.64×10^{2}	1.00×10^{1}	8.41	5.57	9.20×10^{-2}	1.40×10^{-19}		
^{243}Cm	3.90	3.82	3.66	3.43	3.14	4.47×10^{-1}	1.53×10^{-9}			
^{244}Cm	1.26×10^{3}	1.21×10^{3}	1.13×10^{3}	1.00×10^{3}	8.61×10^{2}	2.74×10^{1}	3.32×10^{-14}	3.47×10^{-14}	3.18×10^{-13}	1.52×10^{-12}
^{245}Cm	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.70×10^{-1}	1.58×10^{-1}	7.41×10^{-2}	3.91×10^{-5}	6.60×10^{-38}
^{246}Cm	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.23×10^{-2}	2.83×10^{-2}	7.52×10^{-3}	1.34×10^{-8}	2.50×10^{-31}
^{249}Bk	1.04×10^{-3}	4.63×10^{-4}	9.24×10^{-5}	8.23×10^{-6}						
^{249}Cf	1.78×10^{-6}	3.18×10^{-6}	4.07×10^{-6}	4.25×10^{-6}	4.23×10^{-6}	3.55×10^{-6}	6.03×10^{-7}	1.22×10^{-14}		
^{250}Cf	1.56×10^{-5}	1.48×10^{-5}	1.33×10^{-5}	1.14×10^{-5}	9.21×10^{-6}	7.82×10^{-8}	3.50×10^{-14}	2.45×10^{-14}	6.79×10^{-16}	1.84×10^{-31}
^{252}Cf	1.81×10^{-5}	1.40×10^{-5}	8.27×10^{-6}	3.77×10^{-6}	1.32×10^{-6}					
Total	1.32×10^{5}	1.14×10^{5}	1.01×10^{5}	8.89×10^{4}	7.50×10^{4}	6.74×10^{3}	1.55×10^{3}	4.04×10^{2}	3.36×10^{1}	1.69×10^{1}
Total Thermal Watts	7.86×10^{2}	3.15×10^{2}	1.99×10^{2}	2.00×10^{2}	2.08×10^{2}	1.88×10^{2}	4.98×10^{1}	1.24×10^{1}	9.02×10^{-1}	3.86×10^{-1}

a. Periods are measured from reactor discharge.

TABLE 3.3.12. Actinides and Daughters in Reference High-level Waste as a Function of Decay Time, Uranium-only Recycle, Plutonium Stored Separately (Case 2B)

Isotope	Ci/MTHM for Various Decay Periods[a]								
	0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb(b)				4.57×10^{-10}	4.77×10^{-8}	2.03×10^{-5}	1.09×10^{-3}	8.89×10^{-3}	3.00×10^{-3}
^{226}Ra(c)				3.12×10^{-9}	8.33×10^{-8}	2.03×10^{-5}	1.09×10^{-3}	8.89×10^{-3}	3.00×10^{-3}
^{227}Ac(d)					2.41×10^{-6}	4.13×10^{-6}	2.04×10^{-5}	2.03×10^{-4}	2.81×10^{-4}
^{228}Th(e)	3.99×10^{-3}	1.96×10^{-3}	7.23×10^{-4}	2.08×10^{-4}	5.00×10^{-5}	8.80×10^{-9}	1.13×10^{-9}	1.48×10^{-8}	1.52×10^{-7}
^{229}Th(f)				9.72×10^{-9}	9.55×10^{-7}	9.75×10^{-5}	8.00×10^{-3}	1.70×10^{-1}	4.21×10^{-1}
^{230}Th					4.20×10^{-6}	1.22×10^{-4}	1.41×10^{-3}	8.79×10^{-3}	3.00×10^{-3}
^{231}Th	1.69×10^{-2}	8.47×10^{-5}	8.47×10^{-5}	8.47×10^{-5}	8.49×10^{-5}	8.63×10^{-5}	1.10×10^{-4}	2.66×10^{-4}	2.81×10^{-4}
^{232}Th(g)				7.86×10^{-13}	7.89×10^{-12}	8.22×10^{-11}	1.07×10^{-9}	1.48×10^{-8}	1.52×10^{-7}
^{234}Th	3.15×10^{-1}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}
^{231}Pa	2.37×10^{-6}	2.37×10^{-6}	2.38×10^{-6}	2.39×10^{-6}	2.55×10^{-6}	4.12×10^{-6}	2.04×10^{-5}	2.03×10^{-4}	2.81×10^{-4}
^{233}Pa	4.67×10^{-1}	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.78×10^{-1}	5.29×10^{-1}	5.43×10^{-1}	5.28×10^{-1}	3.94×10^{-1}
^{234m}Pa	3.15×10^{-1}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}
^{234}Pa	3.15×10^{-4}	1.58×10^{-6}	1.58×10^{-3}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}
^{232}U	5.25×10^{-5}	7.94×10^{-5}	9.95×10^{-5}	1.09×10^{-4}	4.87×10^{-5}	8.42×10^{-9}			
^{233}U	1.38×10^{-7}	4.56×10^{-6}	1.12×10^{-5}	2.23×10^{-5}	2.04×10^{-4}	2.16×10^{-3}	2.26×10^{-2}	1.85×10^{-1}	4.18×10^{-1}
^{234}U	1.78×10^{-4}	3.33×10^{-4}	6.06×10^{-4}	1.05×10^{-3}	7.26×10^{-3}	1.73×10^{-2}	1.71×10^{-2}	1.36×10^{-2}	2.54×10^{-3}
^{235}U	8.47×10^{-5}	8.47×10^{-5}	8.47×10^{-5}	8.47×10^{-5}	8.49×10^{-5}	8.63×10^{-5}	1.10×10^{-4}	2.66×10^{-4}	2.81×10^{-4}

a. Years after chemical separation; 1.5 yr elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other activities.

b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.12. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]								
	0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U	1.60×10^{-3}	1.60×10^{-3}	1.60×10^{-3}	1.60×10^{-3}	1.61×10^{-3}	1.75×10^{-3}	2.60×10^{-3}	3.15×10^{-3}	3.07×10^{-3}
^{237}U	1.31×10^{-2}	1.19×10^{-2}	1.04×10^{-2}	8.22×10^{-3}	1.26×10^{-4}	3.95×10^{-6}	1.86×10^{-6}	9.80×10^{-10}	
^{238}U	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-2}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}
^{237}Np	4.67×10^{-1}	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.78×10^{-1}	5.29×10^{-1}	5.43×10^{-1}	5.28×10^{-1}	3.94×10^{-1}
^{239}Np	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{236}Pu	1.80×10^{-3}	1.11×10^{-3}	5.35×10^{-4}	1.58×10^{-4}					
^{238}Pu	1.57×10^{1}	3.20×10^{1}	3.22×10^{1}	3.13×10^{1}	1.89×10^{1}	2.26×10^{-1}	3.35×10^{-19}		
^{239}Pu	1.46	1.46	1.46	1.46	1.50	1.79	3.26	4.46×10^{-1}	1.11×10^{-7}
^{240}Pu	2.26	2.50	2.61	3.29	5.39	4.98	1.98	1.95×10^{-4}	1.17×10^{-9}
^{241}Pu	5.24×10^{2}	4.77×10^{2}	4.55×10^{2}	3.28×10^{2}	5.02	1.58×10^{-1}	7.42×10^{-2}	3.92×10^{-5}	
^{242}Pu	7.91×10^{-3}	7.92×10^{-3}	7.93×10^{-3}	7.95×10^{-3}	8.20×10^{-3}	8.71×10^{-3}	8.83×10^{-3}	7.57×10^{-3}	1.46×10^{-3}
^{241}Am	3.65×10^{2}	3.66×10^{2}	3.66×10^{2}	3.66×10^{2}	3.27×10^{2}	7.77×10^{1}	7.43×10^{-2}	3.92×10^{-5}	
^{242m}Am	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.76	1.12×10^{-1}	1.69×10^{-19}		
^{242}Am	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.76	1.12×10^{-1}	1.69×10^{-19}		
^{243}Am	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{242}Cm	3.48×10^{3}	1.64×10^{2}	1.00×10^{1}	8.37	5.55	9.16×10^{-2}	1.39×10^{-19}		
^{243}Cm	3.82	3.66	3.43	3.07	4.38×10^{-1}	1.50×10^{-9}			
^{244}Cm	1.21×10^{3}	1.13×10^{3}	1.00×10^{3}	8.28×10^{2}	2.64×10^{1}	3.21×10^{-14}	3.47×10^{-14}	3.18×10^{-13}	1.52×10^{-12}
^{245}Cm	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.70×10^{-1}	1.58×10^{-1}	7.41×10^{-2}	3.91×10^{-5}	
^{246}Cm	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.23×10^{-2}	2.83×10^{-2}	7.52×10^{-3}	1.34×10^{-8}	
^{249}Bk	4.63×10^{-4}	9.24×10^{-5}	8.23×10^{-6}	1.46×10^{-7}					
^{249}Cf	3.18×10^{-6}	4.07×10^{-6}	4.25×10^{-6}	4.23×10^{-6}	3.54×10^{-6}	6.02×10^{-7}	1.21×10^{-14}		
^{250}Cf	1.48×10^{-5}	1.33×10^{-5}	1.14×10^{-5}	8.73×10^{-6}	7.42×10^{-8}	3.5×10^{-14}	2.45×10^{-14}	6.79×10^{-16}	
^{252}Cf	1.40×10^{-5}	8.27×10^{-6}	3.77×10^{-6}	1.02×10^{-6}					
Total	5.66×10^{3}	2.22×10^{3}	1.88×10^{3}	1.62×10^{3}	4.32×10^{2}	1.11×10^{2}	1.76×10^{1}	3.17	4.62
Total W/MTHM	1.84×10^{2}	5.95×10^{1}	4.96×10^{1}	4.34×10^{1}	1.34×10^{1}	3.31	3.95×10^{-1}	7.25×10^{-2}	1.10×10^{-1}

a. Years after chemical separation; 1.5 yr elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other activities.

TABLE 3.3.13. Actinides and Daughters in Reference High-level Waste as a Function of Decay Time Uranium-only Recycle, Plutonium Solidified with High-level Waste (Case 2A)

Isotope	Ci/MTHM for Various Decay Periods[a]								
	0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb[b]				8.74×10^{-10}	2.27×10^{-6}	1.46×10^{-3}	7.18×10^{-2}	5.71×10^{-1}	1.04×10^{-1}
^{226}Ra[c]				8.50×10^{-9}	4.59×10^{-6}	1.46×10^{-3}	7.18×10^{-2}	5.71×10^{-1}	1.04×10^{-1}
^{227}Ac[d]					2.42×10^{-6}	7.10×10^{-6}	2.77×10^{-4}	7.20×10^{-3}	1.03×10^{-2}
^{228}Th[e]	3.99×10^{-3}	3.65×10^{-3}	6.74×10^{-3}	1.08×10^{-2}	5.85×10^{-3}	1.01×10^{-6}	2.45×10^{-8}	5.78×10^{-7}	6.26×10^{-6}
^{229}Th[f]				9.72×10^{-9}	9.98×10^{-7}	1.48×10^{-4}	1.77×10^{-2}	3.95×10^{-1}	9.81×10^{-1}
^{230}Th	7.01×10^{-7}	8.57×10^{-7}	1.65×10^{-6}	4.45×10^{-6}	3.02×10^{-4}	8.58×10^{-3}	9.24×10^{-2}	5.66×10^{-1}	1.03×10^{-1}
^{231}Th	1.69×10^{-2}	8.53×10^{-5}	8.62×10^{-5}	8.76×10^{-5}	1.13×10^{-4}	3.66×10^{-4}	2.58×10^{-3}	9.67×10^{-3}	1.03×10^{-2}
^{232}Th[g]				8.18×10^{-13}	1.11×10^{-11}	3.92×10^{-10}	2.45×10^{-8}	5.78×10^{-7}	6.25×10^{-6}
^{234}Th	3.15×10^{-1}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.60×10^{-3}	1.69×10^{-3}
^{231}Pa	2.37×10^{-6}	2.37×10^{-6}	2.38×10^{-6}	2.39×10^{-6}	2.58×10^{-6}	7.10×10^{-6}	2.77×10^{-4}	7.20×10^{-3}	1.03×10^{-2}
^{233}Pa	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.70×10^{-1}	4.63×10^{-1}	1.10	1.26	1.23	9.16×10^{-1}
^{234m}Pa	3.15×10^{-1}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.60×10^{-3}	1.69×10^{-3}
^{234}Pa	3.15×10^{-4}	1.58×10^{-3}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.60×10^{-6}	1.69×10^{-6}
^{232}U	5.32×10^{-5}	5.49×10^{-3}	9.81×10^{-3}	1.22×10^{-2}	5.70×10^{-3}	9.85×10^{-7}			
^{233}U	1.38×10^{-7}	4.56×10^{-6}	1.12×10^{-5}	2.23×10^{-5}	2.20×10^{-4}	3.65×10^{-3}	5.08×10^{-2}	4.32×10^{-1}	9.72×10^{-1}
^{234}U	1.78×10^{-4}	1.77×10^{-2}	4.36×10^{-2}	8.54×10^{-2}	6.16×10^{-1}	1.14	1.11	8.64×10^{-1}	7.07×10^{-2}
^{235}U	8.47×10^{-5}	8.53×10^{-5}	8.62×10^{-5}	8.76×10^{-5}	1.13×10^{-4}	3.66×10^{-4}	2.58×10^{-3}	9.67×10^{-3}	1.03×10^{-2}

a. Years after chemical separation; 1.5 years elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other actinides; separated Pu adds back to waste.

b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.13. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]								
	0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U	1.60×10^{-3}	1.63×10^{-3}	1.66×10^{-3}	1.73×10^{-3}	2.91×10^{-3}	1.41×10^{-2}	8.42×10^{-2}	1.30×10^{-1}	1.27×10^{-1}
^{237}U	1.31×10^{-2}	2.39	2.08	1.64	2.43×10^{-2}	3.95×10^{-6}	1.85×10^{-6}	9.80×10^{-10}	
^{238}U	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.60×10^{-3}	1.69×10^{-3}
^{237}Np	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.70×10^{-1}	5.63×10^{-1}	1.10	1.26	1.23	9.16×10^{-1}
^{239}Np	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{236}Pu	3.61×10^{-1}	2.22×10^{-1}	1.07×10^{-1}	3.17×10^{-2}	9.93×10^{-12}				
^{238}Pu	3.14×10^{3}	3.11×10^{3}	3.04×10^{3}	2.92×10^{3}	1.45×10^{3}	1.52	3.35×10^{-19}		
^{239}Pu	2.92×10^{2}	2.92×10^{2}	2.92×10^{2}	2.92×10^{2}	2.91×10^{2}	2.84×10^{2}	2.22×10^{2}	1.74×10^{1}	1.11×10^{-7}
^{240}Pu	4.52×10^{2}	4.52×10^{2}	4.52×10^{2}	4.53×10^{2}	4.51×10^{2}	4.11×10^{2}	1.63×10^{2}	1.61×10^{-2}	1.17×10^{-9}
^{241}Pu	1.05×10^{5}	9.54×10^{4}	8.29×10^{4}	6.56×10^{4}	9.70×10^{2}	1.58×10^{-1}	7.42×10^{-2}	3.92×10^{-5}	
^{242}Pu	1.58	1.58	1.58	1.58	1.58	1.58	1.55	1.32	2.54×10^{-1}
^{241}Am	3.66×10^{2}	6.84×10^{2}	1.11×10^{3}	1.69×10^{3}	3.44×10^{3}	8.23×10^{2}	7.47×10^{-2}	3.92×10^{-5}	
^{242m}Am	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.76	1.12×10^{-1}	1.69×10^{-19}		
^{242}Am	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.76	1.12×10^{-1}	1.69×10^{-19}		
^{243}Am	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{242}Cm	3.48×10^{3}	1.64×10^{2}	1.00×10^{1}	8.37	5.55	9.16×10^{-2}	1.39×10^{-19}		
^{243}Cm	3.82	3.66	3.43	3.07	4.38×10^{-1}	1.50×10^{-9}			
^{244}Cm	1.21×10^{3}	1.13×10^{3}	1.00×10^{3}	8.28×10^{2}	2.64×10^{1}	3.21×10^{-14}	3.47×10^{-14}	3.18×10^{-13}	1.52×10^{-12}
^{245}Cm	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.70×10^{-1}	1.58×10^{-1}	7.41×10^{-2}	3.91×10^{-5}	
^{246}Cm	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.23×10^{-2}	2.83×10^{-2}	7.52×10^{-3}	1.34×10^{-8}	
^{249}Bk	4.53×10^{-4}	9.24×10^{-5}	8.23×10^{-6}	1.46×10^{-7}					
^{249}Cf	3.18×10^{-6}	4.07×10^{-6}	4.25×10^{-6}	4.23×10^{-6}	3.54×10^{-6}	6.02×10^{-7}	1.21×10^{-14}		
^{250}Cf	1.48×10^{-5}	1.33×10^{-5}	1.14×10^{-5}	8.73×10^{-6}	7.42×10^{-8}	3.50×10^{-14}	2.45×10^{-14}	6.79×10^{-16}	
^{252}Cf	1.40×10^{-5}	8.27×10^{-6}	3.77×10^{-6}	1.02×10^{-6}					
Total	1.14×10^{5}	1.01×10^{5}	8.89×10^{4}	7.19×10^{4}	6.68×10^{3}	1.55×10^{3}	4.03×10^{2}	3.16×10^{1}	1.22×10^{1}
Total W/MTHM	3.15×10^{2}	1.99×10^{2}	2.00×10^{2}	2.09×10^{2}	1.88×10^{2}	4.97×10^{1}	1.23×10^{1}	8.67×10^{-1}	2.91×10^{-1}

a. Years after chemical separation; 1.5 years elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other actinides; separated Pu added back to waste.

TABLE 3.3.14. Actinides and Daughters in Reference High-level Waste as a Function of Decay Time, Uranium and Plutonium Recycle (Case 3)

Element	Ci/MTHM for Various Decay Periods[a]								
	0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb[b]				8.29×10^{-9}	4.62×10^{-7}	6.30×10^{-5}	3.37×10^{-3}	2.72×10^{-2}	6.27×10^{-3}
^{226}Ra[c]				5.56×10^{-8}	6.95×10^{-7}	6.30×10^{-5}	3.37×10^{-3}	2.72×10^{-2}	6.27×10^{-3}
^{227}Ac[d]	9.67×10^{-7}	1.57×10^{-6}	2.41×10^{-6}	3.64×10^{-6}	1.04×10^{-5}	1.22×10^{-5}	2.89×10^{-5}	4.38×10^{-4}	6.51×10^{-4}
^{228}Th[e]	5.30×10^{-3}	2.60×10^{-3}	9.39×10^{-4}	2.43×10^{-4}	4.95×10^{-5}	8.58×10^{-9}	1.88×10^{-9}	3.51×10^{-8}	3.73×10^{-7}
^{229}Th[f]				8.42×10^{-9}	8.29×10^{-7}	8.98×10^{-5}	8.00×10^{-3}	1.74×10^{-1}	4.30×10^{-1}
^{230}Th	1.28×10^{-5}	1.28×10^{-5}	1.29×10^{-5}	1.30×10^{-5}	2.24×10^{-5}	3.75×10^{-4}	4.34×10^{-3}	2.69×10^{-2}	6.25×10^{-3}
^{231}Th	1.59×10^{-2}	7.97×10^{-5}	7.97×10^{-5}	7.97×10^{-5}	7.98×10^{-5}	8.20×10^{-5}	1.40×10^{-4}	6.05×10^{-4}	6.51×10^{-4}
^{232}Th[g]				6.49×10^{-13}	6.59×10^{-12}	7.99×10^{-11}	1.83×10^{-9}	3.50×10^{-8}	3.73×10^{-7}
^{234}Th	3.15×10^{-1}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}
^{231}Pa	1.07×10^{-5}	1.07×10^{-5}	1.07×10^{-5}	1.07×10^{-5}	1.08×10^{-5}	1.22×10^{-5}	2.88×10^{-5}	4.38×10^{-4}	6.51×10^{-4}
^{233}Pa	4.00×10^{-1}	4.01×10^{-1}	4.02×10^{-1}	4.03×10^{-1}	4.22×10^{-1}	5.20×10^{-1}	5.52×10^{-1}	5.39×10^{-1}	4.03×10^{-1}
^{234m}Pa	3.15×10^{-1}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}
^{234}Pa	3.15×10^{-4}	1.57×10^{-6}	1.57×10^{-6}	1.57×10^{-6}	1.57×10^{-6}	1.57×10^{-6}	1.57×10^{-6}	1.57×10^{-6}	1.57×10^{-6}
^{232}U	5.96×10^{-5}	8.28×10^{-5}	1.00×10^{-4}	1.09×10^{-4}	4.81×10^{-5}	8.33×10^{-9}			
^{233}U	1.95×10^{-7}	3.99×10^{-6}	9.70×10^{-6}	1.92×10^{-5}	1.78×10^{-4}	2.03×10^{-3}	2.27×10^{-2}	1.90×10^{-1}	4.28×10^{-1}
^{234}U	1.68×10^{-3}	2.03×10^{-3}	2.67×10^{-3}	3.72×10^{-3}	1.96×10^{-2}	5.33×10^{-2}	5.26×10^{-2}	4.12×10^{-2}	4.75×10^{-3}
^{235}U	7.97×10^{-5}	7.97×10^{-5}	7.97×10^{-5}	7.97×10^{-5}	7.98×10^{-5}	8.20×10^{-5}	1.40×10^{-4}	6.05×10^{-4}	6.51×10^{-4}

a. Years after chemical separation; 1.5 years elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other actinides.

b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.14. (contd)

Element	Ci/MTHM for Various Decay Periods[a]								
	0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U	1.32×10^{-3}	1.32×10^{-3}	1.32×10^{-3}	1.32×10^{-3}	1.37×10^{-3}	1.94×10^{-3}	5.45×10^{-3}	7.75×10^{-3}	7.55×10^{-3}
^{237}U	2.19×10^{-2}	1.99×10^{-2}	1.73×10^{-2}	1.37×10^{-2}	2.47×10^{-4}	4.20×10^{-5}	1.98×10^{-5}	1.04×10^{-8}	
^{238}U	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}	1.57×10^{-3}
^{239}Np	4.73×10^{1}	4.73×10^{1}	4.73×10^{1}	4.73×10^{1}	4.69×10^{1}	4.32×10^{1}	1.91×10^{1}	5.51×10^{-3}	1.71×10^{-6}
^{236}Pu	1.61×10^{-3}	9.91×10^{-4}	4.78×10^{-4}	1.42×10^{-4}					
^{238}Pu	2.77×10^{1}	7.45×10^{1}	7.58×10^{1}	7.44×10^{1}	5.23×10^{1}	9.98×10^{-1}	1.53×10^{-18}		
^{239}Pu	1.79	1.80	1.80	1.81	1.93	3.02	8.90	1.37	1.71×10^{-6}
^{240}Pu	3.66	5.07	6.99	9.74	2.22×10^{1}	2.06×10^{1}	8.20	8.07×10^{-4}	2.48×10^{-8}
^{241}Pu	8.74×10^{2}	7.96×10^{2}	6.92×10^{2}	5.48×10^{2}	9.89	1.68	7.90×10^{-1}	4.17×10^{-4}	
^{242}Pu	1.95×10^{-2}	1.96×10^{-2}	1.96×10^{-2}	1.97×10^{-2}	2.09×10^{-2}	2.35×10^{-2}	2.58×10^{-2}	2.28×10^{-2}	4.39×10^{-3}
^{243}Pu	8.94×10^{-9}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.78×10^{-6}	1.71×10^{-6}
^{241}Am	7.09×10^{2}	7.10×10^{2}	7.10×10^{2}	7.09×10^{2}	6.31×10^{2}	1.51×10^{2}	7.91×10^{-1}	4.17×10^{-4}	
^{242m}Am	4.85×10^{1}	4.81×10^{1}	4.75×10^{1}	4.64×10^{1}	3.08×10^{1}	5.08×10^{-1}	7.69×10^{-19}		
^{242}Am	4.85×10^{1}	4.81×10^{1}	4.75×10^{1}	4.64×10^{1}	3.08×10^{1}	5.08×10^{-1}	7.69×10^{-19}		
^{243}Am	4.73×10^{1}	4.73×10^{1}	4.73×10^{1}	4.73×10^{1}	4.69×10^{1}	4.32×10^{1}	1.91×10^{1}	5.51×10^{-3}	1.71×10^{-6}
^{242}Cm	9.89×10^{3}	4.81×10^{2}	4.31×10^{1}	3.81×10^{1}	2.52×10^{1}	4.17×10^{-1}	6.32×10^{-19}		
^{243}Cm	9.74	9.33	8.74	7.84	1.12	3.83×10^{-9}			
^{244}Cm	7.15×10^{3}	6.62×10^{3}	5.90×10^{3}	4.88×10^{3}	1.55×10^{2}	2.43×10^{-13}	7.36×10^{-13}	6.74×10^{-12}	3.23×10^{-11}
^{245}Cm	1.82	1.82	1.82	1.82	1.81	1.68	7.89×10^{-1}	4.16×10^{-4}	
^{246}Cm	3.41×10^{-1}	3.41×10^{-1}	3.41×10^{-1}	3.41×10^{-1}	3.36×10^{-1}	2.95×10^{-1}	7.84×10^{-2}	1.39×10^{-7}	
^{247}Cm	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.78×10^{-6}	1.71×10^{-6}
^{248}Cm	7.45×10^{-6}	7.45×10^{-6}	7.45×10^{-6}	7.45×10^{-6}	7.45×10^{-6}	7.43×10^{-6}	7.30×10^{-6}	6.12×10^{-6}	1.04×10^{-6}
^{249}Bk	1.34×10^{-2}	2.68×10^{-3}	2.38×10^{-4}	4.23×10^{-6}					
^{249}Cf	9.87×10^{-5}	1.25×10^{-4}	1.30×10^{-4}	1.29×10^{-4}	1.08×10^{-4}	1.84×10^{-5}	3.71×10^{-13}		
^{250}Cf	3.44×10^{-4}	3.10×10^{-4}	2.64×10^{-4}	2.03×10^{-4}	1.72×10^{-6}	4.81×10^{-13}	3.36×10^{-13}	9.34×10^{-15}	
^{251}Cf	3.47×10^{-6}	3.47×10^{-6}	3.46×10^{-6}	3.45×10^{-6}	3.21×10^{-6}	1.61×10^{-6}	1.57×10^{-9}		
^{252}Cf	3.37×10^{-4}	2.00×10^{-4}	9.11×10^{-5}	2.46×10^{-5}					
Total	1.89×10^{4}	8.89×10^{3}	7.63×10^{3}	6.45×10^{3}	1.06×10^{3}	2.68×10^{2}	5.91×10^{1}	4.39	4.77
Total W/MTHM	6.24×10^{2}	2.78×10^{2}	2.37×10^{2}	2.01×10^{2}	3.18×10^{1}	7.53	1.33	1.08×10^{-1}	1.13×10^{-1}

a. Years after chemical separation; 1.5 years elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% U and Pu, but no other actinides.

TABLE 3.3.15. Actinides and Daughters in Reference Spent Fuel as a Function of Decay Time, Uranium and Plutonium Recycle (Case 3)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb(b)					1.3×10^{-8}	7.7×10^{-6}	3.1×10^{-3}	1.5×10^{-1}	1.2	5.0×10^{-1}
^{226}Ra(c)					1.1×10^{-7}	1.4×10^{-5}	3.1×10^{-3}	1.5×10^{-1}	1.2	5.0×10^{-1}
^{227}Ac(d)					3.8×10^{-6}	3.4×10^{-5}	3.5×10^{-4}	3.4×10^{-3}	2.3×10^{-2}	2.9×10^{-2}
^{228}Th(e)	3.1×10^{-3}	5.3×10^{-3}	1.0×10^{-2}	1.6×10^{-2}	2.0×10^{-2}	1.0×10^{-2}	1.7×10^{-6}	1.7×10^{-7}	2.2×10^{-6}	2.3×10^{-5}
^{229}Th(f)					4.3×10^{-8}	1.2×10^{-6}	1.8×10^{-4}	2.4×10^{-2}	5.5×10^{-1}	1.4
^{230}Th	9.9×10^{-6}	1.3×10^{-5}	1.9×10^{-5}	2.9×10^{-5}	4.4×10^{-5}	8.3×10^{-4}	1.8×10^{-2}	1.9×10^{-1}	1.2	5.0×10^{-1}
^{231}Th	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.9×10^{-2}	2.8×10^{-2}	2.9×10^{-2}
^{232}Th(g)					1.3×10^{-10}	1.3×10^{-9}	1.4×10^{-8}	1.7×10^{-7}	2.2×10^{-6}	2.3×10^{-5}
^{234}Th	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{231}Pa	1.0×10^{-5}	1.1×10^{-5}	1.1×10^{-5}	1.2×10^{-5}	1.4×10^{-5}	4.4×10^{-5}	3.5×10^{-4}	3.4×10^{-3}	2.3×10^{-2}	2.9×10^{-2}
^{233}Pa	4.0×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	4.1×10^{-1}	5.6×10^{-1}	1.5	1.8	1.7	1.3
^{234m}Pa	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{234}Pa	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}
^{232}U	8.5×10^{-3}	1.2×10^{-2}	1.7×10^{-2}	2.0×10^{-2}	2.2×10^{-2}	9.7×10^{-3}	1.7×10^{-6}			
^{233}U	3.7×10^{-5}	3.9×10^{-5}	4.3×10^{-5}	4.8×10^{-5}	5.6×10^{-5}	2.4×10^{-4}	4.6×10^{-3}	7.0×10^{-2}	6.0×10^{-1}	1.4
^{234}U	3.2×10^{-1}	3.4×10^{-1}	3.7×10^{-1}	4.1×10^{-1}	4.7×10^{-1}	1.4	2.4	2.3	1.9	4.4×10^{-1}
^{235}U	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.6×10^{-2}	1.9×10^{-2}	2.8×10^{-2}	2.9×10^{-2}

a. Periods are measured from reactor discharge.

b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.15. (contd)

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U	2.6×10^{-1}	2.6×10^{-1}	2.6×10^{-1}	2.6×10^{-1}	2.6×10^{-1}	2.7×10^{-1}	2.9×10^{-1}	4.0×10^{-1}	4.8×10^{-1}	4.6×10^{-1}
^{237}U	4.6	4.4	4.0	3.5	2.9	4.3×10^{-2}	4.2×10^{-5}	2.0×10^{-5}	1.0×10^{-8}	
^{238}U	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{237}Np	4.0×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	4.1×10^{-1}	5.6×10^{-1}	1.5	1.8	1.7	1.3
^{239}Np	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.3×10^{1}	1.9×10^{1}	5.5×10^{-3}	1.7×10^{-6}
^{236}Pu	4.1×10^{-1}	3.2×10^{-1}	1.9×10^{-1}	9.6×10^{-2}	3.6×10^{-2}					
^{238}Pu	5.4×10^{3}	5.5×10^{3}	5.5×10^{3}	5.4×10^{3}	5.2×10^{3}	2.6×10^{3}	3.3			
^{239}Pu	3.6×10^{2}	3.6×10^{2}	3.6×10^{2}	3.6×10^{2}	3.6×10^{2}	3.6×10^{2}	3.5×10^{2}	2.8×10^{2}	2.2×10^{1}	1.7×10^{-6}
^{240}Pu	7.3×10^{2}	7.3×10^{2}	7.3×10^{2}	7.4×10^{2}	7.4×10^{2}	7.4×10^{2}	6.8×10^{2}	2.7×10^{2}	2.7×10^{-2}	2.5×10^{-8}
^{241}Pu	1.8×10^{5}	1.8×10^{5}	1.6×10^{5}	1.4×10^{5}	1.2×10^{5}	1.7×10^{3}	1.7	7.9×10^{-1}	4.2×10^{-4}	
^{242}Pu	3.9	3.9	3.9	3.9	3.9	3.9	3.9	3.8	3.3	6.3×10^{-1}
^{243}Pu	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.7×10^{-6}
^{241}Am	4.2×10^{2}	7.1×10^{2}	1.2×10^{3}	2.0×10^{3}	2.7×10^{3}	5.8×10^{3}	1.4×10^{3}	7.9×10^{-1}	4.2×10^{-4}	
^{242m}Am	4.9×10^{1}	4.9×10^{1}	4.8×10^{1}	4.8×10^{1}	4.6×10^{1}	3.1×10^{1}	5.1×10^{-1}			
^{242}Am	4.9×10^{1}	4.9×10^{1}	4.8×10^{1}	4.8×10^{1}	4.6×10^{1}	3.1×10^{1}	5.1×10^{-1}			
^{243}Am	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.7×10^{1}	4.3×10^{1}	1.9×10^{1}	5.5×10^{-3}	1.7×10^{-6}
^{242}Cm	4.7×10^{4}	9.9×10^{3}	4.8×10^{2}	4.3×10^{1}	3.8×10^{1}	2.5×10^{1}	4.2×10^{-1}			
^{243}Cm	9.9	9.7	9.3	8.7	8.0	1.1	3.9×10^{-9}			
^{244}Cm	7.4×10^{3}	7.2×10^{3}	6.6×10^{3}	5.9×10^{3}	5.1×10^{3}	1.6×10^{2}				
^{245}Cm	1.8	1.8	1.8	1.8	1.8	1.8	1.7	7.9×10^{-1}	4.2×10^{-4}	
^{246}Cm	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.4×10^{-1}	3.0×10^{-1}	7.8×10^{-2}	1.4×10^{-7}	
^{247}Cm	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}	1.7×10^{-6}
^{248}Cm	7.4×10^{-6}	7.5×10^{-6}	7.4×10^{-6}	7.4×10^{-6}	7.5×10^{-6}	7.5×10^{-6}	7.4×10^{-6}	7.3×10^{-6}	6.1×10^{-6}	1.0×10^{-6}
^{249}Bk	3.0×10^{-2}	1.3×10^{-2}	2.7×10^{-3}	2.4×10^{-4}	9.5×10^{-6}					
^{249}Cf	5.8×10^{-5}	9.9×10^{-5}	1.2×10^{-4}	1.3×10^{-4}	1.3×10^{-4}	1.1×10^{-4}	1.8×10^{-5}			
^{250}Cf	3.6×10^{-4}	3.4×10^{-4}	3.1×10^{-4}	2.6×10^{-4}	2.1×10^{-4}	1.8×10^{-6}				
^{251}Cf	3.5×10^{-6}	3.5×10^{-6}	3.5×10^{-6}	3.5×10^{-6}	3.5×10^{-6}	3.2×10^{-6}	1.6×10^{-6}	1.6×10^{-9}		
^{252}Cf	4.4×10^{-4}	3.4×10^{-4}	2.0×10^{-4}	9.1×10^{-5}	3.2×10^{-5}					
Total	2.4×10^{5}	2.0×10^{5}	1.7×10^{5}	1.5×10^{5}	1.3×10^{5}	1.2×10^{4}	2.5×10^{3}	6.0×10^{2}	5.0×10^{1}	2.3×10^{1}
Total Thermal Watts	2.2×10^{3}	8.7×10^{2}	5.2×10^{2}	4.9×10^{2}	4.8×10^{2}	3.2×10^{2}	8.1×10^{1}	1.8×10^{1}	1.3	5.2×10^{-1}

a. Periods are measured from reactor discharge.

TABLE 3.3.16. Activity in Reference Plutonium Stored at a Repository as a Function of Decay Time, Uranium-Only Recycle, Plutonium Separated and Stored Separately (Case 2B)

Isotope	Ci/kg PuO_2[a,b] for Various Decay Periods[c]								
	1.5 yr	2.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb(d)				4.6×10^{-11}	2.4×10^{-7}	1.6×10^{-4}	7.8×10^{-3}	6.2×10^{-2}	1.1×10^{-2}
^{226}Ra(e)				5.9×10^{-10}	5.0×10^{-7}	1.6×10^{-4}	7.8×10^{3}	6.2×10^{-2}	1.1×10^{-2}
^{227}Ac(f)				3.3×10^{-12}	1.9×10^{-9}	3.3×10^{-7}	2.8×10^{-5}	7.7×10^{-4}	1.1×10^{-3}
^{228}Th(g)		5.6×10^{-5}	6.6×10^{-4}	1.1×10^{-3}	6.4×10^{-4}	1.1×10^{-7}	2.5×10^{-9}	6.2×10^{-8}	6.7×10^{-7}
^{229}Th(h)				9.1×10^{-13}	4.8×10^{-9}	5.5×10^{-4}	1.1×10^{-3}	2.5×10^{-2}	6.1×10^{-2}
^{230}Th		4.2×10^{-9}	1.0×10^{-7}	3.3×10^{-7}	3.3×10^{-5}	9.3×10^{-4}	1.0×10^{-2}	6.2×10^{-2}	1.1×10^{-2}
^{231}Th		3.1×10^{-8}	1.6×10^{-7}	2.8×10^{-7}	3.1×10^{-6}	3.1×10^{-5}	2.7×10^{-4}	1.0×10^{-3}	1.1×10^{-3}
^{232}Th(i)				3.5×10^{-15}	3.5×10^{-13}	3.4×10^{-11}	2.6×10^{-9}	6.2×10^{-8}	6.7×10^{-7}
^{233}Pa		2.2×10^{-6}	6.9×10^{-5}	2.1×10^{-4}	9.3×10^{-3}	6.3×10^{-2}	7.9×10^{-2}	7.7×10^{-2}	5.7×10^{-2}
^{232}U		3.4×10^{-4}	6.1×10^{-3}	1.3×10^{-3}	6.3×10^{-4}	1.1×10^{-7}			
^{233}U					1.8×10^{-6}	1.6×10^{-4}	3.1×10^{-3}	2.7×10^{-2}	6.0×10^{-2}
^{234}U		9.6×10^{-4}	4.7×10^{-3}	8.4×10^{-3}	6.7×10^{-2}	1.2×10^{-1}	1.2×10^{-1}	9.3×10^{-2}	7.5×10^{-3}
^{235}U		3.1×10^{-8}	1.6×10^{-7}	2.8×10^{-7}	3.1×10^{-6}	3.1×10^{-5}	2.7×10^{-4}	1.0×10^{-3}	1.1×10^{-3}

a. Radioactivities of isotopes other than Pu are all derived from the decay of the Pu.

b. Radon (and daughters) assumed to remain with parent material.

c. Periods are measured from reactor discharge, separation after 1.5 years assumed.

d. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

e. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

f. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

g. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

h. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th

i. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.16. (contd)

Isotope	Ci/kg PuO_2[a,b] for Various Decay Periods[c]								
	1.5 yr	2.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U		1.4×10^{-6}	7.2×10^{-6}	1.3×10^{-5}	1.4×10^{-4}	1.3×10^{-3}	9.0×10^{-3}	1.4×10^{-2}	1.3×10^{-2}
^{237}U		2.7×10^{-1}	2.3×10^{-1}	1.9×10^{-1}	2.6×10^{-3}	1.3×10^{-21}			
^{237}NP		3.0×10^{-6}	6.9×10^{-5}	2.1×10^{-4}	9.3×10^{-3}	6.3×10^{-2}	7.9×10^{-2}	7.7×10^{-2}	5.7×10^{-2}
^{236}Pu	4.0×10^{-2}	3.1×10^{-2}	1.2×10^{-2}	4.4×10^{-3}	1.1×10^{-12}				
^{238}Pu	3.4×10^{2}	3.4×10^{2}	3.3×10^{2}	3.2×10^{2}	1.5×10^{2}	1.4×10^{-1}			
^{239}Pu	3.2×10^{1}	3.2×10^{1}	3.2×10^{1}	3.2×10^{1}	3.2×10^{1}	3.1×10^{1}	2.4×10^{1}	1.9	1.4×10^{-11}
^{240}Pu	4.9×10^{1}	4.9×10^{1}	4.9×10^{1}	4.9×10^{1}	4.9×10^{1}	4.5×10^{1}	1.8×10^{1}	1.8×10^{-3}	5.3×10^{-16}
^{241}Pu	1.1×10^{4}	1.1×10^{4}	9.1×10^{3}	7.6×10^{3}	1.1×10^{2}	5.3×10^{-17}			
^{242}Pu	1.8×10^{-1}	1.8×10^{-1}	1.7×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}	1.4×10^{-1}	2.7×10^{-2}
^{241}Am		1.8×10^{1}	8.1×10^{1}	1.3×10^{2}	3.4×10^{2}	8.2×10^{1}	4.5×10^{-5}		
Total	1.2×10^{4}	1.1×10^{4}	9.6×10^{3}	8.0×10^{3}	6.9×10^{2}	1.5×10^{2}	4.3×10^{1}	3.1	8.4×10^{-1}
Total W/kg PuO	1.4×10^{1}	1.5×10^{1}	1.5×10^{1}	1.7×10^{1}	1.8×10^{1}	4.8	1.3	8.3×10^{-2}	1.9×10^{-2}

a. Radioactivities of isotopes other than Pu are all derived from the decay of the Pu.

b. Radon (and daughters) assumed to remain with parent material.

c. Periods are measured from reactor discharge, separation ofter 1.5 years assumed.

TABLE 3.3.17. Activity in Reference Plutonium Separated at Fuel Reprocesssing Plant as a Function of Decay Time, Uranium and Plutonium Recycle (Case 3)

	Ci/MTHM[a] for Various Decay Periods[b]							
Isotope	1.5 yr	2.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb[c]			7.3×10^{-10}	3.9×10^{-6}	2.5×10^{-3}	1.2×10^{-1}	9.9×10^{-1}	1.8×10^{-1}
^{226}Ra[d]			9.5×10^{-9}	8.0×10^{-6}	2.5×10^{-3}	1.2×10^{-1}	9.9×10^{-1}	1.8×10^{-1}
^{227}Ac[e]			3.7×10^{-11}	2.1×10^{-8}	3.7×10^{-6}	3.2×10^{-4}	8.6×10^{-3}	1.2×10^{-2}
^{228}Th[f]		4.7×10^{-4}	9.7×10^{-3}	5.3×10^{-3}	9.1×10^{-7}	3.8×10^{-8}	9.2×10^{-7}	9.9×10^{-6}
^{229}Th[g]			1.4×10^{-11}	7.2×10^{-8}	8.4×10^{-5}	1.6×10^{-2}	3.7×10^{-1}	9.3×10^{-1}
^{230}Th		6.7×10^{-8}	6.5×10^{-6}	5.2×10^{-4}	1.5×10^{-2}	1.6×10^{-1}	9.8×10^{-1}	1.8×10^{-1}
^{231}Th		3.5×10^{-7}	3.5×10^{-6}	3.5×10^{-5}	3.4×10^{-4}	3.0×10^{-3}	1.2×10^{-2}	1.2×10^{-2}
^{232}Th[h]			5.2×10^{-14}	5.2×10^{-12}	5.0×10^{-10}	3.8×10^{-8}	9.1×10^{-7}	9.9×10^{-6}
^{233}Pa		4.6×10^{-5}	3.9×10^{-3}	1.4×10^{-1}	9.5×10^{-1}	1.2	1.2	8.7×10^{-1}
^{232}U		2.8×10^{-3}	1.1×10^{-2}	5.1×10^{-3}	8.9×10^{-7}			
^{233}U				2.6×10^{-5}	2.5×10^{-3}	4.7×10^{-2}	4.1×10^{-1}	9.2×10^{-1}
^{234}U		1.5×10^{-2}	1.5×10^{-1}	1.1	2.0	1.9	1.5	1.2×10^{-1}
^{235}U		3.5×10^{-7}	3.5×10^{-6}	3.5×10^{-5}	3.4×10^{-4}	3.0×10^{-3}	1.2×10^{-2}	1.2×10^{-2}

a. Radioactivity in one MTHM spent fuel. Radioactivities of isotopes other than Pu are all derived from the decay of the Pu.

b. Periods are measured from reactor discharge.

c. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

d. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

e. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

f. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

g. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

h. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.17. (contd)

Isotope	Ci/MTHM[a] for Various Decay Periods[b]							
	1.5 yr	2.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U		2.1×10^{-5}	2.1×10^{-4}	2.1×10^{-3}	2.0×10^{-2}	1.3×10^{-1}	2.1×10^{-1}	2.0×10^{-1}
^{237}U		4.1	2.7	4.0×10^{-2}				
^{237}Np		4.6×10^{-5}	3.9×10^{-3}	1.4×10^{-1}	9.5×10^{-1}	1.2	1.2	8.7×10^{-1}
^{236}Pu	3.3×10^{-1}	2.6×10^{-1}	2.9×10^{-2}					
^{238}Pu	5.5×10^3	5.5×10^3	5.1×10^3	2.5×10^3	2.3			
^{239}Pu	3.6×10^2	3.6×10^2	3.6×10^2	3.6×10^2	3.5×10^2	2.7×10^2	2.1×10^1	
^{240}Pu	7.3×10^2	7.3×10^2	7.3×10^2	7.2×10^2	6.6×10^2	2.6×10^2	2.6×10^{-2}	
^{241}Pu	1.7×10^5	1.7×10^5	1.1×10^5	1.6×10^3				
^{242}Pu	3.9	3.9	3.9	3.9	3.9	3.8	3.2	6.2×10^{-1}
^{241}Am		2.7×10^2	2.2×10^3	5.2×10^3	1.2×10^3	6.9×10^{-4}		
Total	1.8×10^5	1.7×10^5	1.2×10^5	1.0×10^4	2.3×10^3	5.4×10^2	4.2×10^1	4.9
Total Thermal watts	2.2×10^2	2.3×10^2	2.8×10^2	2.9×10^2	7.3×10^1	1.7×10^1	8.7×10^{-1}	8.6×10^{-2}

a. Radioactivity in one MTHM spent fuel. Radioactivities of isotopes other than Pu are all derived from the decay of the Pu.

b. Periods are measured from reactor dishcarge.

TABLE 3.3.18. Radionuclide Inventory in Reference Mixed Oxide Fuel as a Function of Decay Time (Case 3)

Isotope	Ci/MTHM in MOX Fuel for Various Decay Periods[a,b]							
	0 yr[a]	1 yr[c]	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb[d]		7.0×10^{-12}	8.9×10^{-9}	1.7×10^{-5}	9.5×10^{-3}	4.6×10^{-1}	3.7	9.4×10^{-1}
^{226}Ra[e]		6.6×10^{-10}	9.5×10^{-8}	3.4×10^{-5}	9.5×10^{-3}	4.6×10^{-1}	3.7	9.4×10^{-1}
^{227}Ac[f]		4.9×10^{-9}	4.5×10^{-7}	2.2×10^{-5}	3.2×10^{-4}	3.9×10^{-3}	4.3×10^{-2}	5.8×10^{-2}
^{228}Th[g]		1.6×10^{-3}	3.4×10^{-2}	1.9×10^{-2}	3.2×10^{-6}	1.3×10^{-7}	3.2×10^{-6}	3.5×10^{-5}
^{229}Th[h]			5.4×10^{-11}	2.5×10^{-7}	3.0×10^{-4}	5.6×10^{-2}	1.3	3.3
^{230}Th		3.1×10^{-6}	5.2×10^{-5}	2.1×10^{-3}	5.5×10^{-2}	5.9×10^{-1}	3.7	9.4×10^{-1}
^{231}Th		1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.6×10^{-2}	2.5×10^{-2}	5.5×10^{-2}	5.8×10^{-2}
^{232}Th[i]			1.9×10^{-13}	1.9×10^{-11}	1.8×10^{-9}	1.4×10^{-7}	3.3×10^{-6}	3.5×10^{-5}
^{234}Th		3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{231}Pa		3.1×10^{-7}	3.1×10^{-6}	3.1×10^{-5}	3.2×10^{-4}	3.9×10^{-3}	4.3×10^{-2}	5.8×10^{-2}
^{233}Pa		1.6×10^{-4}	1.4×10^{-2}	5.0×10^{-1}	3.4	4.2	4.1	3.1
^{234m}Pa		3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{234}Pa		3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}	3.2×10^{-4}
^{232}U		9.8×10^{-3}	3.9×10^{-2}	1.8×10^{-2}	3.1×10^{-6}			
^{233}U		2.3×10^{-10}	2.2×10^{-7}	9.2×10^{-5}	8.8×10^{-3}	1.7×10^{-1}	1.4	3.3
^{234}U	3.3×10^{-1}	3.9×10^{-1}	8.6×10^{-1}	4.1	7.3	7.1	5.6	7.4×10^{-1}
^{235}U	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.6×10^{-2}	2.5×10^{-2}	5.5×10^{-2}	5.8×10^{-2}

a. Based on decay of reference Pu purified 1.5 years after reactor discharge and of freshly purified natural U.

b. Radon (and daughters) assumed to remain with parent material.

c. Time assumed for reference MOX fuel fabrication.

d. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

e. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

f. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

g. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th, ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

h. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th, ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

i. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 3.3.18. (contd)

Isotope	Ci/MTHM in MOX Fuel for Various Decay Periods[a,b]							
	0 yr[a]	1 yr[c]	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{236}U		7.5×10^{-5}	7.5×10^{-4}	7.4×10^{-3}	7.1×10^{-2}	4.7×10^{-1}	7.3×10^{-1}	7.1×10^{-1}
^{237}U		1.5×10^{1}	9.6	1.4×10^{-1}	7.1×10^{-20}			
^{238}U	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}	3.2×10^{-1}
^{237}Np		1.6×10^{-4}	1.4×10^{-2}	5.0×10^{-1}	3.4	4.2	4.1	3.1
^{236}Pu	1.2	9.0×10^{-1}	1.0×10^{-1}	3.2×10^{-11}				
^{238}Pu	1.9×10^{4}	1.9×10^{4}	1.8×10^{4}	8.9×10^{3}	8.1			
^{239}Pu	1.3×10^{3}	1.3×10^{3}	1.3×10^{3}	1.3×10^{3}	1.2×10^{-3}	9.5×10^{2}	7.4×10^{1}	5.8×10^{-10}
^{240}Pu	2.6×10^{3}	2.6×10^{3}	2.6×10^{3}	2.6×10^{3}	2.3×10^{3}	9.2×10^{2}	9.1×10^{-2}	
^{241}Pu	6.1×10^{5}	5.8×10^{5}	3.8×10^{5}	5.7×10^{3}	2.8×10^{-15}			
^{242}Pu	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.3×10^{1}	1.1×10^{1}	2.2
^{241}Am		9.6×10^{2}	7.8×10^{3}	1.8×10^{4}	4.4×10^{3}	2.4×10^{-3}		
Total	6.4×10^{5}	6.1×10^{5}	4.1×10^{5}	3.7×10^{4}	8.0×10^{3}	1.9×10^{3}	1.5×10^{2}	5.0×10^{1}
Total W/MTHM	7.8×10^{2}	8.0×10^{2}	9.8×10^{2}	1.0×10^{3}	2.6×10^{2}	6.0×10^{1}	4.1	1.2

a. Based on decay of reference Pu purified 1.5 years after reactor discharge and of freshly purified natural U.

b. Radon (and daughters) assumed to remain with parent material.

c. Time assumed for reference Mox fuel fabrication.

WASTE CHARACTERIZATION TABLES

TABLE 3.3.19. Compactable and Combustible Wastes from the Nuclear Power Plant

Source	Components, wt%		Density, kg/m^3	Volume, m^3/GWe-yr	Radionuclide Content	
					Activation Products[a]	Fission Products[a]
HEPA filters	Glass	40	1.6×10^2	5	5×10^{-3}	8×10^{-4}
	Wood	60				
Combustible trash	Paper	23	1.2×10^2	2×10^2	7×10^{-4}	7×10^{-4}
	Plastic	67				
	Rubber	6				
	Wood	3				
	Cloth	1				

a. Given as fraction of Ci values listed in Table 3.3.4. To obtain Ci/yr for a given isotope, multiply Ci/yr given for that isotope in Table 3.3.4 by the fraction listed here.

TABLE 3.3.20. Concentrated Liquids, Wet Wastes, and Particulate Solids from the Nuclear Power Plant

Source	Components, wt%		Density, kg/m^3	Volume, m^3/GWe-yr	Radionuclide Content		
						Activation Products[a]	Fission Products[a]
Bead resins	Polystyrene	50	7.2×10^2	2.9×10^1	^3H	4×10^{-2}	9.7×10^{-1}
	H_2O	50			All others	1.8×10^{-1}	
Powdered resins	Polystyrene	50	6.7×10^2	2.9×10^1	^3H	3×10^{-2}	2×10^{-2}
	H_2O	50			All others	4×10^{-1}	
Filter precoat sludge	Filter aids	40	4.3×10^2	3.4×10^1	^3H	3×10^{-2}	3×10^{-3}
	H_2O	60			All others	1×10^{-3}	
Cartridge filters	Filter media	80	5×10^2	5	^3H	2×10^{-3}	5×10^{-4}
	H_2O	20			All others	4.2×10^{-1}	
Sulfate concentrate	Na_2SO_4	25	1.21×10^3	2.6×10^2	^3H	9×10^{-1}	4×10^{-3}
	H_2O	75			All others	8×10^{-4}	
Borate concentrate	$Na_2B_4O_7$	10	1.04×10^3	4	^3H	1×10^{-2}	1×10^{-3}
	H_2O	90			All others	2×10^{-4}	

a. Given as fraction of Ci values listed in Table 3.3.4. To obtain Ci/yr for a given isotope, multiply Ci/yr given for that isotope in Table 3.3.4 by the fraction listed here.

TABLE 3.3.21. Failed Equipment and Noncompactable, Noncombustible Waste from the Nuclear Power Plant

Source	Components, wt%		Density, kg/m^3	Volume, m^3/GWe-yr	Radionuclide Content		
					Activation Products[a]		Fission Products[a]
Noncombustible trash	Metal Glass	90 10	2.5×10^2	4.0×10^1	2×10^{-4}		2×10^{-4}
Reactor scrap/ failed equipment	Ferrous alloys	100	5×10^2	6	^{3}H All others	5.6 2.3×10^2	3×10^{-4}

a. Given as fraction of Ci values listed in Table 3.3.4. To obtain Ci/yr for a given isotope, multiply Ci/yr given for that isotope in Table 3.3.4 by the fraction listed here.

TABLE 3.3.22. Spent Fuel from the Nuclear Power Plant, Once-Through Fuel Cycle

Source	Components, wt%		Weight, MTHM/GWe-yr[a]	Radionuclide Content		
				Activation Products[b]	Fission Products[c]	Actinides[d]
PWR	Core	79.5	35	1.0	1.0	1.0
	Zircaloy	16.5				
	Hardware	4				
BWR	Core	75.4	42	1.0	1.0	1.0
	Zircaloy	21.0				
	Hardware	3.6				
Average LWR	Core	78	38	1.0	1.0	1.0
	Zircaloy	18				
	Hardware	4				

a. GWe-yr of energy produced by a 2:1 ratio of PWR's and BWR's based on average exposure.
b. Given as fraction of Ci values listed in Tables 3.3.6 and 3.3.7. Add the two values to obtain the total activation products.
c. Given as fraction of Ci values listed in Table 3.3.8.
d. Given as fraction of Ci values listed in Table 3.3.10.

TABLE 3.3.23. Gaseous Waste from the Independent Spent Fuel Storage Basin

Source	Components, wt%		Density kg/m^3	Operations	Volume	Radionuclide Content Fission Products[a]	
HVAC	Air	100	1.22	Receiving	5×10^5 m^3/MTHM	All	0
				Storing	1×10^6 m^3/MTHM-yr	^{3}H	1×10^{-6}(a)
						^{85}Kr	7×10^{-7}(a)
						Other	0
				Shipping	5×10^5 m^3/MTHM	All	0
				Composite[b]	1.2×10^6 m^3/MTHM-yr	^{3}H	1×10^{-6}
						^{85}Kr	7×10^{-7}
						Other	0
Vaporized excess water	H_2O	100	6×10^{-1}	Receiving	5×10^2 m^3/MTHM[c]	^{14}C	3×10^{-8}(d)
						^{129}I	1×10^{-7}(d)
						Particulates	1×10^{-5}(e)
				Storing	2×10^3 m^3/MTHM-yr[c]	^{14}C	1×10^{-8}(a)
						^{129}I	9×10^{-9}(a)
						Particulates	7×10^{-6}(e)
				Shipping	5×10^3 m^3/MTHM[c]	^{14}C	1×10^{-8}(d)
						^{129}I	3×10^{-8}(d)
						Particulates	3×10^{-6}(e)
				Composite[b]	2×10^3 m^3/MTHM-yr[c]	^{14}C	1×10^{-8}
						^{129}I	3×10^{-8}
						Particulates	9×10^{-6}
Process off-gas	Air	100	1.22	Receiving	7×10^3 m^3/MTHM	^{3}H	12×10^{-6}(d)
						^{14}C	3×10^{-6}(d)
						^{85}Kr	6×10^{-5}(d)
						^{129}I	1×10^{-5}(d)
						Other	0
				Composite[b]	1×10^3 m^3/MTHM-yr	^{3}H	3×10^{-7}
						^{14}C	5×10^{-7}
						^{85}Kr	1×10^{-5}
						^{129}I	2×10^{-6}
						Other	0

a. Given as fraction per year of Ci values listed in Table 3.3.8 or 3.3.9. Use 3.5 year values.
b. Composite based on operating mode in which one-sixth of storage inventory is shipped out and replaced each year.
c. Volume of water vapor at 100 C.
d. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9. Use 0.5 year values for receiving and 6.5 year values for shipping.
e. Given as fraction of Ci values listed for corresponding operation in Table 3.3.4. Apply to activation products as well as fission products.

TABLE 3.3.24. Compactable and Combustible Wastes from the Independent Spent Fuel Storage Basin

Source	Components, wt%		Density, kg/m^3	Operation	Volume	Radionuclide Content Activation Products[a]	Fission Products[a]
Combustible Trash	Cellulosic	50	1.2×10^2	Receiving	6×10^{-1} m^3/MTHM	7×10^{-3}	7×10^{-3}
	PVC	15		Storing	8×10^{-2} m^3/MTHM-yr	7×10^{-3}	7×10^{-3}
	Polystyrene	15		Shipping	1.6×10^{-1} m^3/MTHM	7×10^{-3}	7×10^{-3}
	Latex	10		Composite[b]	2.1×10^{-1} m^3/MTHM-yr	7×10^{-3}	7×10^{-3}
	Neoprene	10					
Ventilation filters	Glass	40	1.6×10^2	Receiving	8.5×10^{-3} m^3/MTHM	1×10^{-4}	1×10^{-4}
	Wood	60		Storing	5.3×10^{-3} m^3/MTHM-yr	1×10^{-4}	1×10^{-4}
				Shipping	5.3×10^{-3} m^3/MTHM	1×10^{-4}	1×10^{-4}
				Composite[b]	$7.6 \times .0^{-3}$ m^3/MTHM-yr	1×10^{-4}	1×10^{-4}

a. Given as fraction of Ci values listed for corresponding operation in Table 3.3.4.
b. Composite based on operating mode in which one-sixth of storage inventory is shipped out and replaced each year.

TABLE 3.3.25. Concentrated Liquids, Wet Wastes and Particulate Solids from the Independent Spent Fuel Storage Basin

Source	Components, wt%		Density, kg/m^3	Operation	Volume	Radionuclide Content: Activation Products[a]	Radionuclide Content: Fission Products[a]
Bead resins	Polystyrene	50	7.2×10^2	Receiving	2×10^{-3} m^3/MTHM	3×10^{-3}	3×10^{-3}
	Water	50		Storing	3×10^{-4} m^3/MTHM-yr	3×10^{-3}	3×10^{-3}
				Shipping	6×10^{-4} m^3/MTHM	3×10^{-3}	3×10^{-3}
				Composite[b]	7.3×10^{-4} m^3/MTHM-yr	3×10^{-3}	3×10^{-3}
Filter precoat sludge	Cellulosic	20	4.3×10^2	Receiving	7×10^{-3} m^3/MTHM	3×10^{-1}	3×10^{-1}
	Diatomaceous earth	20		Storing	1×10^{-3} m^3/MTHM-yr	3×10^{-1}	3×10^{-1}
				Shipping	2×10^{-3} m^3/MTHM	3×10^{-1}	3×10^{-1}
	Water	60		Composite[b]	2.5×10^{-3} m^3/MTHM-yr	3×10^{-1}	3×10^{-1}
Sulfate concentrate	Na_2SO_4	25	1.2×10^3	Receiving	5×10^{-3} m^3/MTHM	3×10^{-1}	3×10^{-1}
	Water	75		Storing	1×10^{-3} m^3/MTHM-yr	3×10^{-1}	3×10^{-1}
				Shipping	3×10^{-3} m^3/MTHM	3×10^{-1}	3×10^{-1}
				Composite[b]	2.3×10^{-3} m^3/MTHM-yr	3×10^{-1}	3×10^{-1}
Miscellaneous solution concentrates	Decontamination agents and corrosion products	25	1.2×10^3	Receiving	1×10^{-2} m^3/MTHM	4×10^{-1}	4×10^{-1}
				Storing	1×10^{-3} m^3/MTHM-yr	4×10^{-1}	4×10^{-1}
				Shipping	3×10^{-3} m^3/MTHM	4×10^{-1}	4×10^{-1}
	Water	75		Composite[b]	3.2×10^{-3} m^3/MTHM-yr	4×10^{-1}	4×10^{-1}

a. Given as fraction of Ci values listed for corresponding operation in Table 3.3.4.
b. Composite based on operating mode in which one-sixth of storage inventory is shipped out and replaced each year.

TABLE 3.3.26. Failed Equipment and Noncompactable, Noncombustible Waste from the Independent Spent Fuel Storage Basin

Source	Components, wt%		Density, kg/m^3	Operation	Volume	Radionuclide Content: Activation Products[a]	Radionuclide Content: Fission Products[a]
Noncombustible trash	Metal	90	3×10^2	Receiving	5×10^{-2} m^3/MTHM	6×10^{-4}	6×10^{-4}
	Glass	100		Storing	7×10^{-3} m^3/MTHM-yr	6×10^{-4}	6×10^{-4}
				Shipping	1×10^{-2} m^3/MTHM	6×10^{-4}	6×10^{-4}
				Composite[b]	1.7×10^{-2} m^3/MTHM-yr	6×10^{-4}	6×10^{-4}
Failed equipment	Ferrous alloys	100	5×10^2	Receiving	7×10^{-3} m^3/MTHM	8×10^{-6}	8×10^{-6}
				Storing	4×10^{-3} m^3/MTHM-yr	8×10^{-6}	8×10^{-6}
				Shipping	7×10^{-3} m^3/MTHM	8×10^{-6}	8×10^{-6}
				Composite[b]	6.3×10^{-3} m^3/MTHM-yr	8×10^{-6}	8×10^{-6}

a. Given as fraction of Ci values listed for corresponding operation in Table 3.3.4.
b. Composite based on operating mode in which one-sixth of storage inventory is shipped out and replaced each year.

TABLE 3.3.27. Gaseous Waste from the Fuel Reprocessing Plant

Source	Components, wt%	Density, kg/m³	Volume, m³/MTHM	Radionuclide Content: Fission Products(a)		Actinides(b)	
Fuel shear and dissolver off-gas	Air 94.2	1.22	6.8×10^{2}	^{3}H	5×10^{-2}	1×10^{-7}	
	CO_2 0.07			^{14}C	1.0		
	NO_x 3.2			^{85}Kr	1.0		
	Xe 0.6			^{129}I	9.9×10^{-1}		
	H_2O 1.9			^{106}Ru	2×10^{-4}		
				All others	1×10^{-7}		
Vessel off-gas	Air 100	1.22	3×10^{4}	^{3}H	1×10^{-3}	1×10^{-7}	
				^{85}Kr	1×10^{-6}		
				^{129}I	5×10^{-3}		
				All others	1×10^{-7}		
Main building HVAC	Air 100	1.22	1.8×10^{6}		1×10^{-11}	1×10^{-11}	
Vaporized excess water	H_2O 100	6×10^{-1}	2×10^{4}(c)	^{3}H	7.2×10^{-1}	U	1×10^{-11}
				^{129}I	1×10^{-5}	Pu	1×10^{-11}
				^{106}Ru	1×10^{-10}	All others	1×10^{-16}
				^{95}Zr	1×10^{-11}		
				^{95}Nb	1×10^{-12}		
				All others	1×10^{-16}		
UF_6 plant process off-gas	Air 98.7	1.22	7×10^{4}	^{95}Zr	3×10^{-12}	U	1×10^{-5}
	CO_2 0.05			^{95}Nb	3×10^{-12}	Pu	1×10^{-11}
	NO_x 0.0015			^{106}Ru	4×10^{-12}	All others	0
	H_2 0.003			All others	0		
	HF 9×10^{-5}						
	H_2S 1.5×10^{-3}						
	H_2O 1.3						
Storage basin HVAC	Air 99	1.22	1.3×10^{6}	0		0	
	H_2O 1.0						

a. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9.
b. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.
c. Volume of water vapor at 100°C.

TABLE 3.3.28. Hulls and Assembly Hardware from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Weight, kg/MTHM	Radionuclide Content		
					Activation Products	Fission Products(c)	Actinides(d)
Hardware	Fe	57	1×10^3	5.6×10^1	1.0(a)	0	0
	Ni	17					
	Cr	17					
	Cu	5					
	Other metals	4					
Fuel residue (hulls)	Zr	97.7	1×10^3	2.66×10^2	1.0(b)	^3H 1.5×10^{-1}	5×10^{-4}
	Sn	1.6				^{85}Kr 0	
	Other metals	0.7				^{129}I 0	
						All others 5×10^{-4}	

a. Given as fraction of Ci values listed in Table 3.3.6.
b. Given as fraction of Ci values listed in Table 3.3.7.
c. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9.
d. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.

TABLE 3.3.29. High-level Liquid Waste from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Volume, m^3/MTHM	Radionuclide Content			
					Fission Products(a)		Actinides(b)	
Raffinate from first cycle extractor and dissolver sludge	Nitrate salts	15	1.2×10^3	6×10^{-1}	^3H	8×10^{-2}	U	5×10^{-3}
	HNO_3	10			^{85}Kr	0	Pu	5×10^{-3}
	H_2O	75			^{129}I	5×10^{-3}	All others	1.0
					All others	1.0		

a. Given as fraction of Ci values listed in Tables 3.3.8 or 3.3.9.
b. Given as fraction of Ci values listed in Tables 3.3.11 or 3.3.15. Alternatively the actinide content may be obtained directly from Tables 3.3.12, 3.3.13 or 3.3.14.

TABLE 3.3.30. Plutonium Waste from the Fuel Reprocessing Plant, Uranium-Only Recycle

Source	Components, wt%		Density, kg/m³	Volume, m³/MTHM	Radionuclide Content per kg PuO_2			
					Fission Products[a]		Actinides[b]	
Pu nitrate solution	$Pu(NO_3)_4$	34	1.4×10^3	3.6×10^{-2}	^3H	0		1.0
	HNO_3	9			^{85}Kr	0		
	H_2O	57			^{129}I	0		
or					All others	1×10^{-9}		
Powder	PuO_2	100	5×10^3	1.9×10^{-3}	^3H	0		1.0
					^{85}Kr	0		
					^{129}I	0		
					All others	1×10^{-9}		

a. Given as fraction of Ci values listed in Table 3.3.8.
b. Given as fraction of Ci values listed in Table 3.3.16.

TABLE 3.3.31. Intermediate-level Concentrated Liquids, Wet Wastes, and Particulate Solids from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m³	Volume, m³/MTHM	Radionuclide Content			
					Fission Products[a]		Actinides[b]	
Main plant ILLW (TRU)	$NaNO_3$	18	1.25×10^3	1.1×10^{-1}	^3H	1×10^{-3}	U	1×10^{-3}
	Na_3PO_4	8			^{85}Kr	0	Pu	1×10^{-3}
	Misc. salts	4			Zr	1×10^{-3}	All others	1×10^{-5}
	H_2O	70			Nb	1×10^{-3}		
					Ru	1×10^{-3}		
					I	3×10^{-3}		
					All others	1×10^{-5}		
Main plant silica gel (TRU)	SiO_2	70	8×10^2	5×10^{-3}	Zr	1×10^{-7}	Pu	1×10^{-7}
	H_2O	30			Nb	1×10^{-7}	All others	0
					All others	0		
Storage basin bead resins (non-TRU)	Resin	50	7.2×10^2	2×10^{-3}	3×10^{-3}[c]		0	
	H_2O	50						
Storage basin filter precoat sludge (non-TRU)	Sludge	40	4.3×10^2	7×10^{-3}	3×10^{-1}[c]		0	
	H_2O	60						
Storage basin sulfate concentrate (non-TRU)	Na_2SO_4	25	1.21×10^3	5×10^{-3}	3×10^{-1}[c]		0	
	H_2O	75						
Storage basin miscellaneous solutions (non-TRU)	Salts	25	1.2×10^3	1×10^{-2}	4×10^{-1}[c]		0	
	H_2O	75						

a. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9.
b. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.
c. Given as fraction of Ci values listed in Table 3.3.4. Include activation products. To obtain Ci/yr for a given isotope, multiply Ci/yr given for that isotope in Table 3.3.4 by the fraction listed here.

TABLE 3.3.32. Low-level Concentrated Liquids, Wet Wastes, and Particulate Solids from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content			
					Fission Products(a)		Actinides(b)	
UF_6 plant fluorinator bed residues (TRU)	Al_2O_3	100	1.5×10^3	4×10^{-3}	^3H	0	U	6×10^{-6}
					^{85}Kr	0	Pu	2×10^{-7}
					^{129}I	0	All others	0
					Zr	1×10^{-8}		
					Nb	1×10^{-8}		
					Ru	1×10^{-7}		
					All others	0		
UF_6 plant fluorinator fines (TRU)	Al_2O_3	50	1.5×10^3	3×10^{-2}	Zr	1×10^{-7}	U	1.5×10^{-3}
	CaF_2	46			Nb	1×10^{-7}	Pu	1×10^{-6}
	Uranium fluorides	4			Ru	1×10^{-6}	All others	0
					All others	0		
UF_6 plant K_2UO_4 mud	KOH	3	1.2×10^3	7×10^{-2}	0		U	1.3×10^{-3}
	KF	4					All others	0
	K_2UO_4	3						
	$Fe(OH)_2$	18						
	Misc. salts	1						
	H_2O	71						
UF_6 plant waste dryer discharge(c)	CaF_2	62	5×10^2	5×10^{-1}	0			0
	$Ca(OH)_2$	15						
	$CaSO_4$	8						
	KOH	12						
	KF	2						
	Misc. salts	1						

a. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9.
b. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.
c. Treated as nonradioactive waste.

TABLE 3.3.33. Intermediate-level Compactable and Combustible Wastes from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content: Fission Products[a]		Radionuclide Content: Actinides[b]	
Storage basin combustible trash (non-TRU)	Paper/rags	40	1.2×10^2	6×10^{-1}	7×10^{-3}[c]		0	
	PVC	20						
	Neoprene	11						
	Polyethylene	14						
	Latex	11						
	Wood	4						
Main plant combustible trash (TRU)	Same as above		1.2×10^2	4×10^{-1}	1×10^{-6}		1×10^{-6}	
PuO_2 conversion combustible trash (TRU)	Same as above		1.2×10^2	3×10^{-2}	0		5×10^{-4}[d]	
UF_6 plant combustible trash (non-TRU)	Same as above		1.2×10^2	5×10^{-2}	0		U	5×10^{-4}
							All others	0
Storage basin ventilation filters (non-TRU)	Wood	60	1.6×10^2	1×10^{-2}	7×10^{-3}[c]		0	
	Glass	40						
Main plant ventilation filters (TRU)	Same as above		1.6×10^2	1.4×10^{-1}	1×10^{-5}		1×10^{-5}	
UF_6 plant ventilation filters (non-TRU)	Same as above		1.6×10^2	5×10^{-3}	0		U	1×10^{-5}
							All others	0
PuO_2 conversion ventilation filters (TRU)	Metal	60	1.6×10^2	2×10^{-2}	0		2×10^{-3}[d]	
	Glass	40						
IX bead resins (TRU)	Polystyrene	50	7.2×10^2	5×10^{-3}	^{129}I	2×10^{-3}	U	1×10^{-5}
	H_2O	50			Zr	1×10^{-5}	Pu	1×10^{-5}
					Nb	1×10^{-5}	All others	1×10^{-7}
					Ru	1×10^{-5}		
					All others	1×10^{-7}		
Degraded extractant (TRU)	TBP	30 (vol)	8×10^2	8×10^{-3}	^{129}I	1×10^{-4}	Pu	1×10^{-4}
	Dodecane	70 (vol)			Zr, Nb, Ru	1×10^{-6}		
					All others	1×10^{-10}	All others	1×10^{-6}

a. Given as fraction of Ci values listed in Table 3.3.8 pr 3.3.9, exclude the volatile isotopes ^3H and ^{85}Kr, also ^{129}I except as noted.
b. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.
c. Given as fraction of Ci values listed in Table 3.3.4. Include activation products. To obtain Ci/yr for a given isotope, multiply Ci/yr given for that isotope in Table 3.3.4 by the fraction listed here.
d. Given as fraction of Ci values listed in Table 3.3.17.

TABLE 3.3.34. Low-level Compactable and Combustible Wastes from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content		
					Fission Products(a)		Actinides(b)
Main plant combustible trash (TRU)	Paper/rags	40	1.2×10^2	1.2	^3H	0	1×10^{-9}
	PVC	20			^{85}Kr	0	
	Neoprene	11			^{129}I	0	
	Polyethylene	14			All others	1×10^{-9}	
	Latex	11					
	Wood	4					

a. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9.
b. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.

TABLE 3.3.35. Transuranic Failed Equipment and Noncompactable, Noncombustible Trash from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content		
					Fission Products(a)		Actinides(b)
Main plant noncombustible trash	Ferrous metal	90	2.5×10^2	4×10^{-1}	1×10^{-6}		1×10^{-6}
	Glass	10					
UF_6 plant noncombustible trash	Same as above		2.5×10^2	1×10^{-2}	0	U	1×10^{-4}
						All others	0
PuO_2 conversion noncombustible trash	Same as above		2.5×10^2	6×10^{-3}	0		1×10^{-4}(c)
Main plant failed equipment	Ferrous alloys	100	5×10^2	2×10^{-1}	1×10^{-6}		1×10^{-6}
UF_6 plant failed equipment	Same as above		5×10^2	1×10^{-2}	0	U	1×10^{-5}
						All others	0
PuO_2 conversion failed equipment	Same as above		5×10^2	2×10^{-2}	0		1×10^{-4}(c)

a. Given as fraction of Ci values listed in Table 3.3.8 or 3.3.9, exclude the volatile isotopes ^3H, ^{85}Kr, and ^{129}I.
b. Given as fraction of Ci values listed in Table 3.3.11 or 3.3.15.
c. Given as fraction of Ci values listed in Table 3.3.17.

TABLE 3.3.36. Nontransuranic Failed Equipment and Noncompactable, Noncombustible Waste from the Fuel Reprocessing Plant

Source	Components, wt%		Density, kg/m^3	Volume, m^3/MTHM	Radionuclide Content: Activation Products[a]	Radionuclide Content: Fission Products[a]
Storage basin non-combustible trash	Ferrous metal	90	2.5×10^2	5×10^{-2}	6×10^{-4}	6×10^{-4}
	Glass	10				
Storage basin failed equipment	Ferrous metal	100	5×10^2	1×10^{-2}	8×10^{-6}	8×10^{-6}

a. Given as fraction of Ci values listed in Table 3.3.4. To obtain Ci/yr for a given isotope, multiply Ci/yr given for that isotope in Table 3.3.4 by the fraction listed here.

TABLE 3.3.37. Gaseous Waste from the Mixed Oxide Fuel Fabrication Plant

Source	Components, wt%	Density, kg/m^3	Volume, m^3/MTHM	Radionuclide Content Actinides[a]	
Influent to air filtration system	Air	1.22	1.1×10^6		3×10^{-7}
Vaporized excess water	H_2O	6×10^{-1}	2×10^{3}[b]	Am	1×10^{-10}
				All others	1×10^{-12}

a. Given as fraction of Ci values listed in Table 3.3.18.
b. Volume of water vapor at 100°C.

TABLE 3.3.38. Compactable and Combustible Wastes from the Mixed Oxide Fuel Fabrication Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content Actinides(a)
HEPA filters	Metal	60	1.6×10^2	1×10^{-1}	7×10^{-4}
	Glass	40			
Combustible trash	PVC	33	1.2×10^2	5×10^{-1}	3×10^{-4}
	Cellulosics	30			
	Polyethylene	18			
	Latex	9			
	Neoprene	9			
	Styrene	1			

a. Given as fraction of Ci values listed in Table 3.3.18.

TABLE 3.3.39. Concentrated Liquids, Wet Wastes, and Particulate Solids from the Mixed Oxide Fuel Fabrication Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content Actinides(a)	
Process solutions	$Al(NO_3)_3$	16	1.2×10^3	2×10^{-2}		1×10^{-5}
	$Ca(NO_3)_2$	6				
	CaF_2	0.2				
	H_2O	78				
Scrap recovery solutions	$Al(NO_3)_3$	2	1.2×10^3	3.5×10^{-1}	Am	2×10^{-2}
	$Ca(NO_3)_2$	11			All others	1×10^{-4}
	AlF_3	0.4				
	NH_4NO_3	7				
	$NaNO_3$	7				
	H_2O	73				

a. Given as fraction of Ci listed in Table 3.3.18.

TABLE 3.3.40. Failed Equipment and Noncompactable, Noncombustible Waste from the Mixed Oxide Fuel Fabrication Plant

Source	Components, wt%		Density, kg/m^3	Volume, $m^3/MTHM$	Radionuclide Content Actinides[a]
Noncombustible trash	Metal	90	2.5×10^2	2×10^{-1}	1×10^{-4}
	Glass	10			
Failed equipment	Metal	80	1×10^3	2×10^{-1}	1×10^{-5}
	Insulating brick	20			

a. Given as fraction of Ci values listed in Table 3.3.18.

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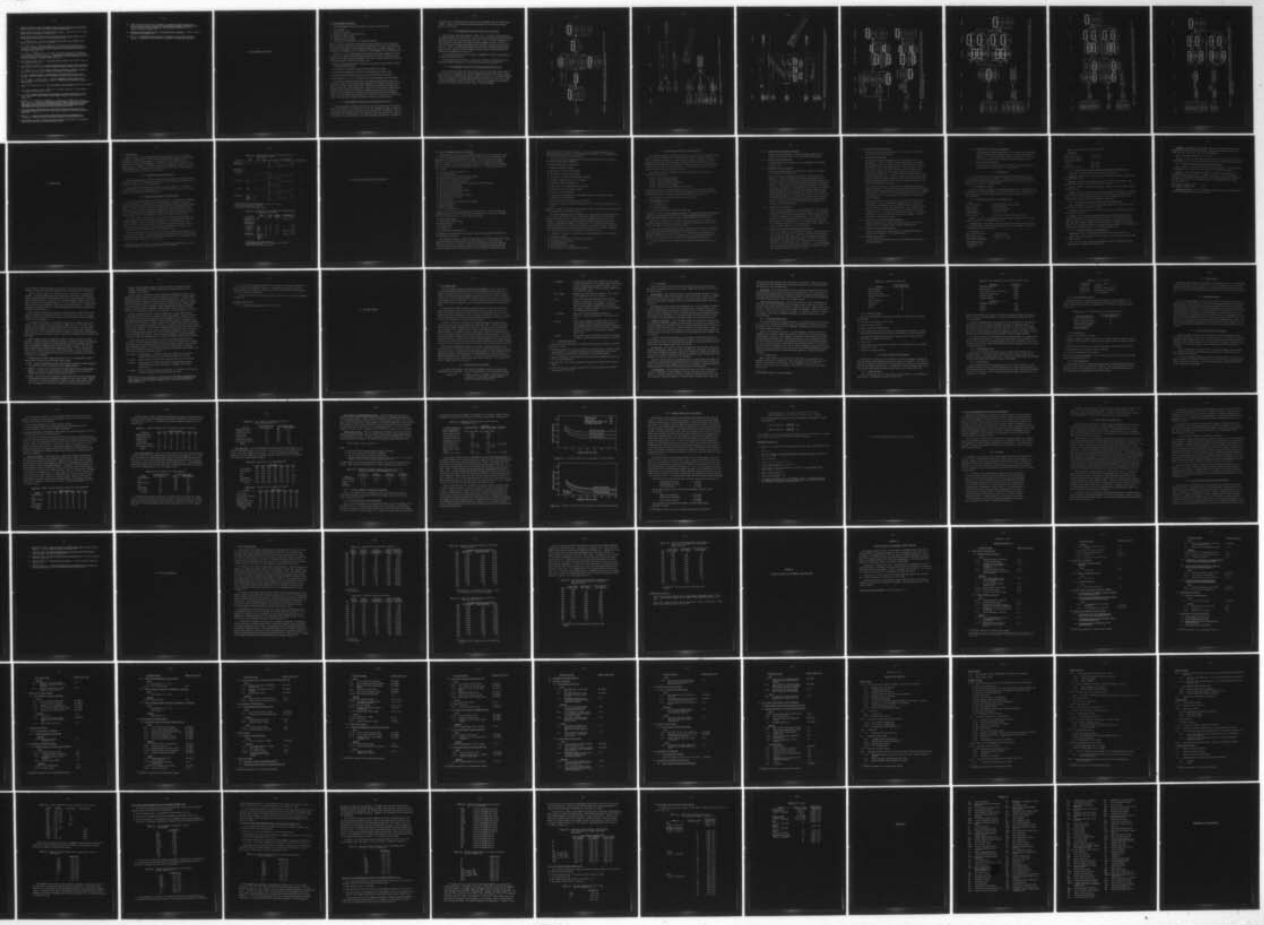

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3.4 WASTE MANAGEMENT ALTERNATIVES

3.4 WASTE MANAGEMENT ALTERNATIVES

The waste management system has potentially six steps for each waste type.

1. waste treatment
2. onsite interim storage
3. transport to central site interim storage
4. central site interim storage
5. transport to final isolation
6. final isolation (assumed here to be a geologic repository).

Steps 2, 3, and 4 are only required in the event a geologic repository is not available at the time the waste is treated. One additional step required in the case of the once-through fuel cycle is lag storage to permit radioactivity decay prior to treatment (packaging). Lag storage prior to treatment is also an alternative for HLW management in any reprocessing cycle.

Figures 3.4.1 - 3.4.9 show management alternatives that have been selected for detailed analysis in this report and identify the tables where each waste type is described and the sections of this report where each alternative is discussed. Aside from providing an overview of the major sections to follow, these figures show:

- how the primary wastes described in Section 3.3 are combined as feed streams to alternative waste treatment options
- how treatment of airborne and gaseous effluents generate solid wastes
- how each waste management step relates to the preceding and succeeding steps
- how the alternative technology options apply to different fuel cycle options.

For most waste types, at least 2 technologies for each waste management step are analyzed.

To facilitate the description and analysis of complete waste management systems, one alternative at each step was selected to be a component of an integrated system early in the study. In some cases, further analysis during the study has indicated that another alternative might have been a more optimum choice. Selection of these reference alternatives was necessary to fully account for the effects of secondary waste quantities and radioactive releases to the environment and to develop subsystem costs and complete waste management system costs. The chosen reference alternative is indicated in the figures by a heavy black outline. The combination of reference alternatives make up the reference system.

3.4.1 Waste Management Alternatives for the Once-Through Fuel Cycle

The waste management alternatives evaluated for the once-through fuel cycle are shown in Figures 3.4.1 and 3.4.2. The alternatives for spent fuel management are shown in Figure 3.4.1. The secondary wastes produced in the steps are all considered to be nontransuranic (non-TRU) wastes and are not considered further in this analysis. Other than the spent fuel itself, the only wastes from the once-through fuel cycle whose management requirements are considered in

this report are the airborne and gaseous wastes from the independent spent fuel storage basins (ISFSBs). Figure 3.4.2 shows the waste treatment components for the airborne and gaseous wastes at a large ISFSB.

3.4.2 Waste Management Alternatives for Other Fuel Cycle Options

Except for some of the waste components, Figures 3.4.3 - 3.4.8 apply to both uranium-only and uranium-plutonium recycle fuel cycles. Figure 3.4.3 shows alternatives for managing airborne and gaseous wastes from a fuel reprocessing plant (FRP). Figure 3.4.4 shows alternatives for fuel residue (hulls and hardware) and high-level waste management at an FRP. Figure 3.4.5 shows alternatives for managing concentrated liquids, wet wastes, and particulate solid wastes at an FRP. Figure 3.4.6 shows waste management alternatives for compactable and combustible waste at an FRP. Figure 3.4.7 shows the alternatives for managing failed equipment and non-combustible, noncompactable waste at an FRP.

The alternatives shown in Figure 3.4.8 apply to management of the FRP plutonium product in the case of uranium-only recycle.

The alternatives shown in Figure 3.4.9 apply only in the case of plutonium recycle and show waste treatment alternatives at a mixed oxide fuel fabrication plant (MOX FFP).

3.4.3 Waste Management Alternatives for Decommissioning Retired Facilities

At the end of their useful life, the facilities used in the light water reactor (LWR) fuel cycle options will be decommissioned. Because decommissioning wastes differ from the primary wastes, which have already been described only in quantity and not in kind, the waste management proced res developed for the primary wastes generated during operation of facilities can also be used during decommissioning of the facilities. Details of the decommissioning procedures are given in Section 8.0, along with a description of the wastes produced.

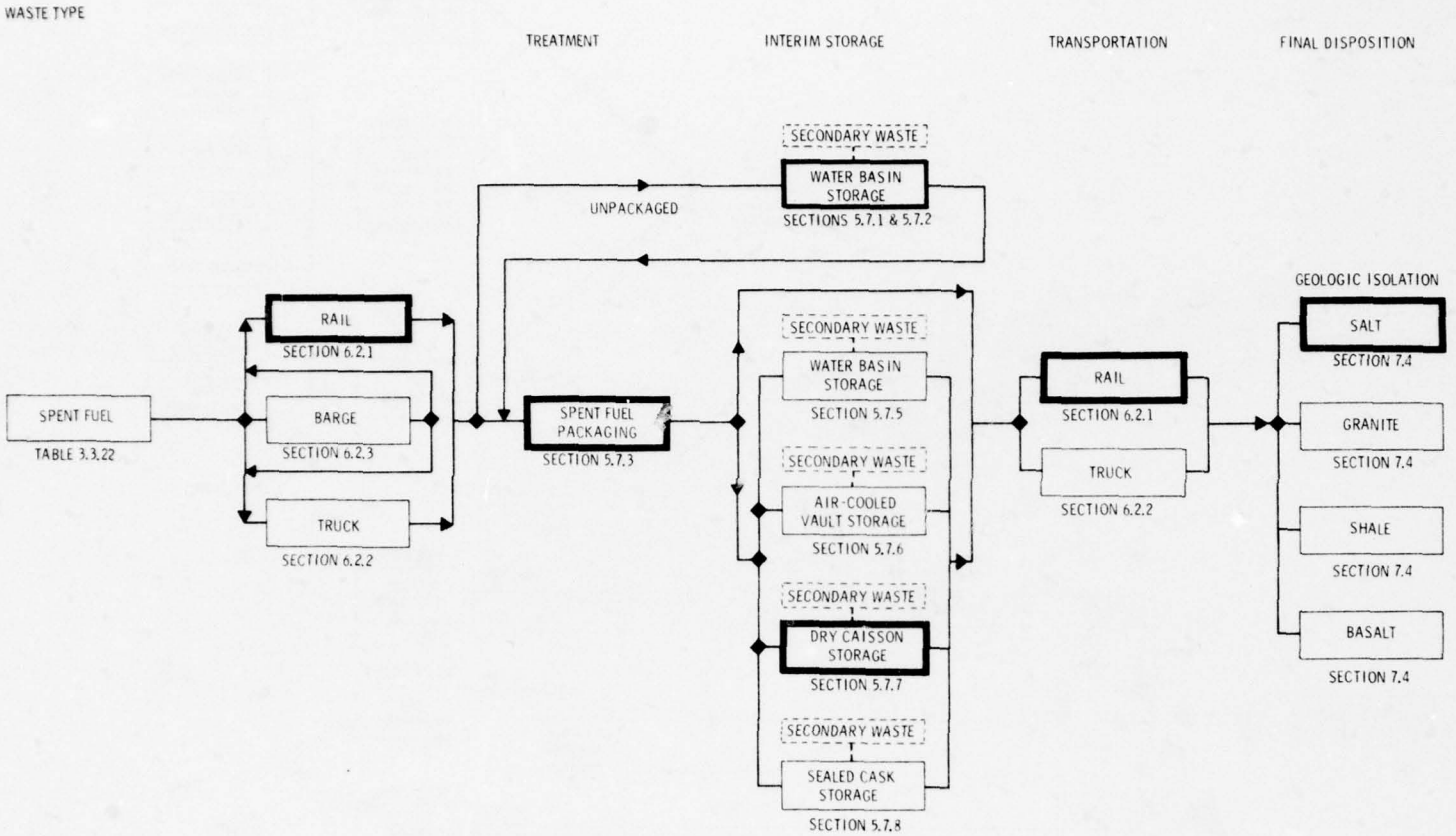

FIGURE 3.4.1. Waste Management Alternatives for Spent Fuel in the Once-Through Cycle (Heavy outlines show alternatives selected for the reference waste management system.)

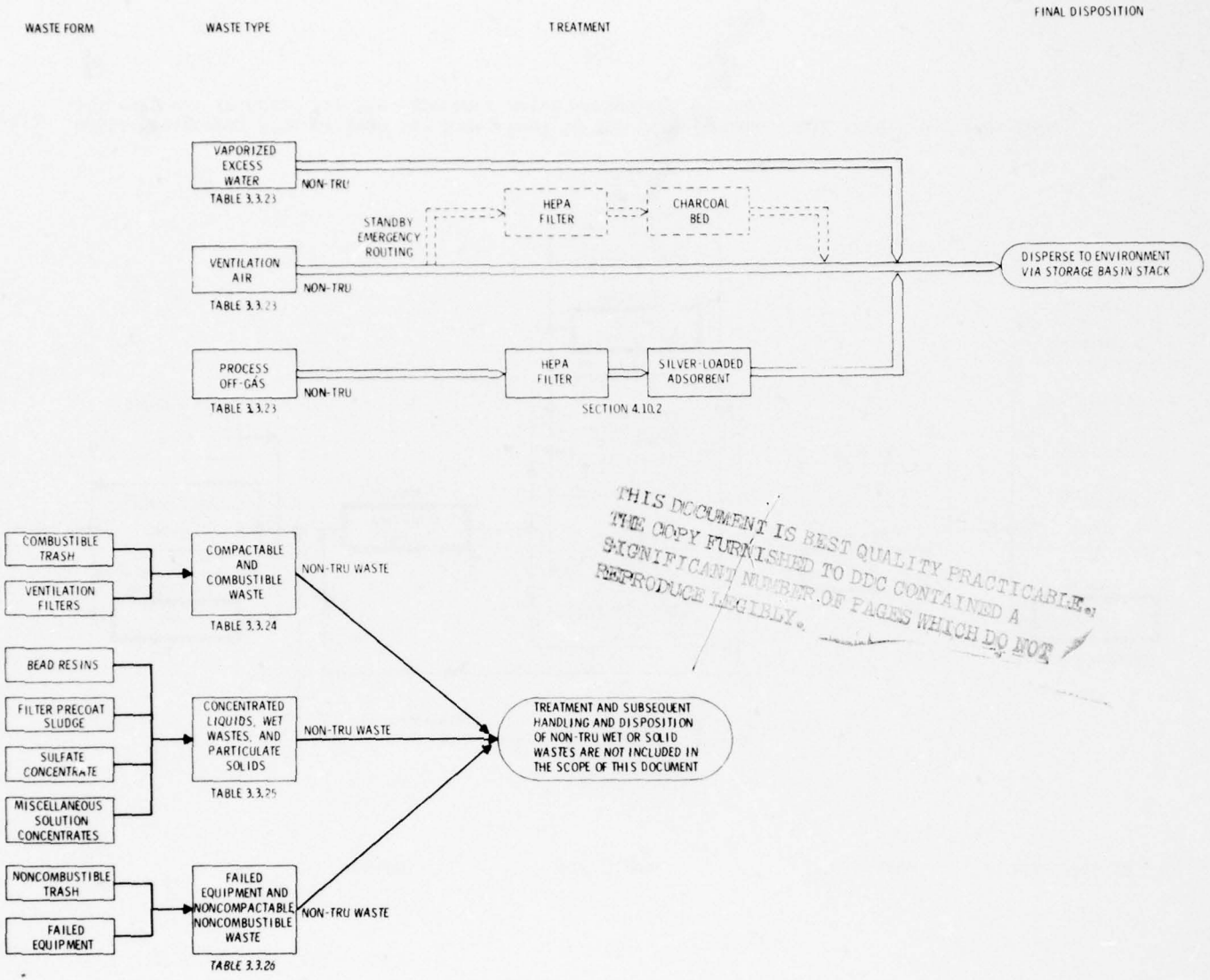

FIGURE 3.4.2. Waste Management Alternatives for an Independent Spent Fuel Storage Basin

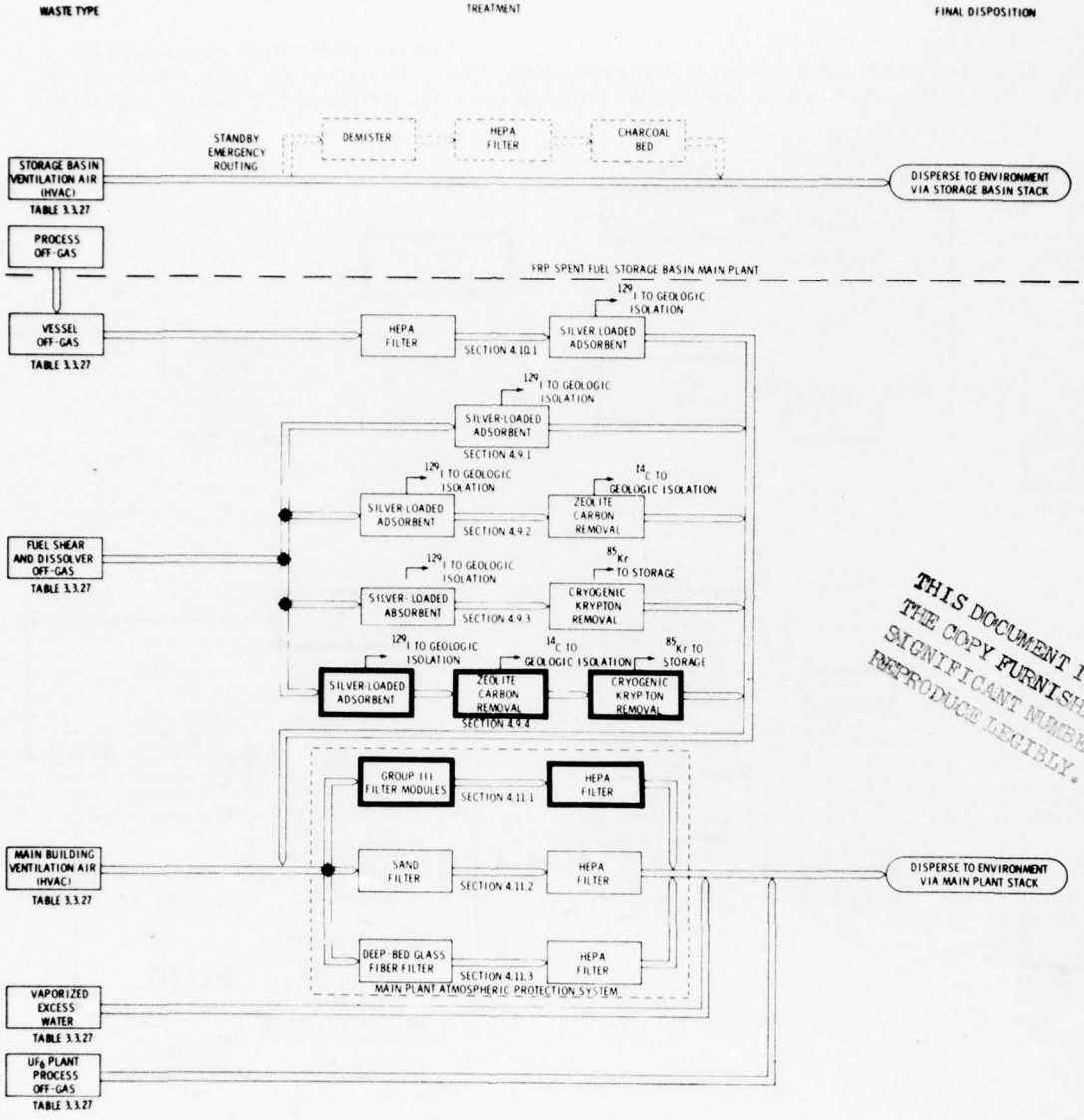

FIGURE 3.4.3. Waste Management Alternatives for Airborne and Gaseous Wastes from a Fuel Reprocessing Plant (Heavy outlines show alternatives selected for the reference waste management system.)

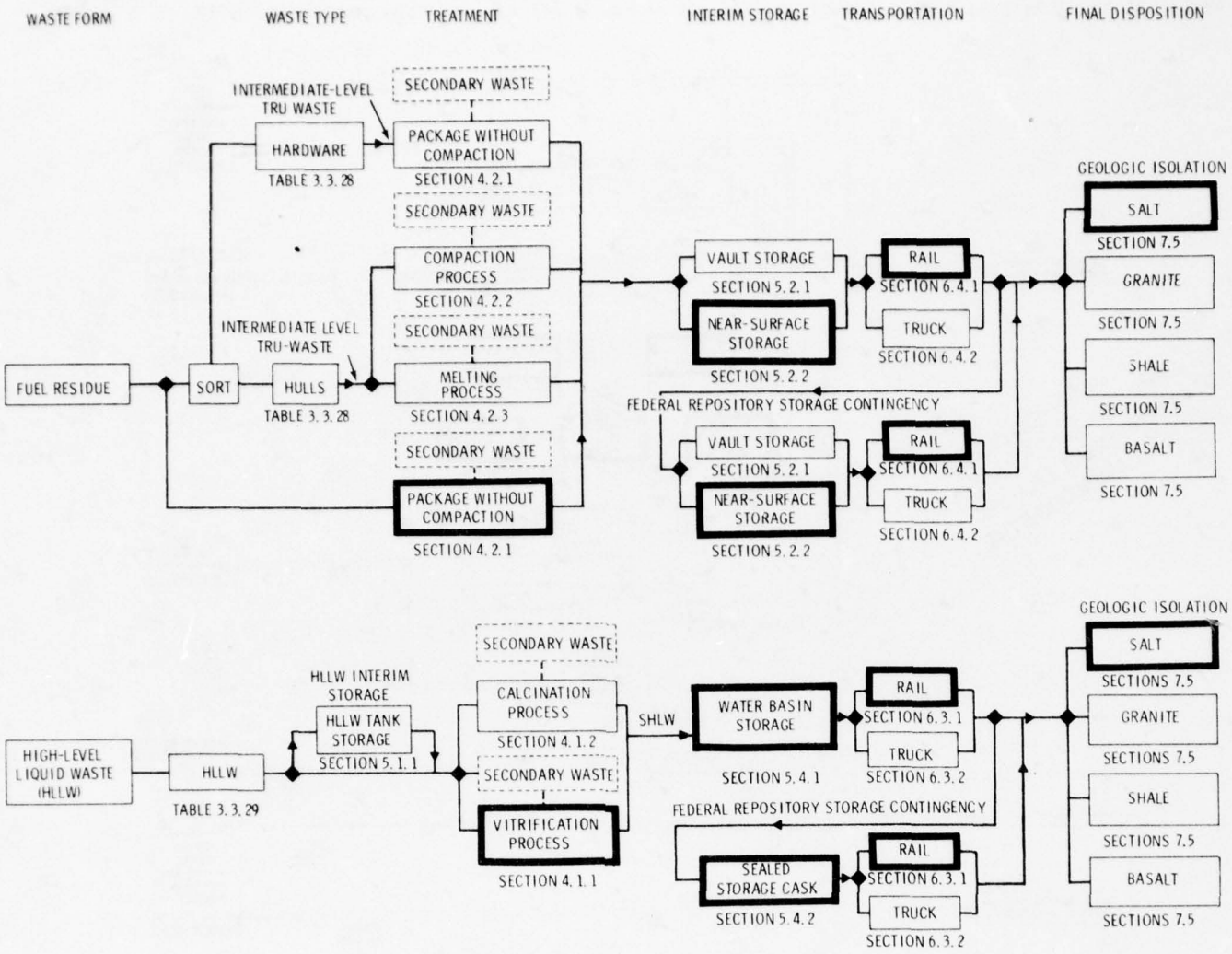

FIGURE 3.4.4. Waste Management Alternatives for Fuel Residue (Hulls and Assembly Hardware) and High-Level Waste from a Fuel Reprocessing Plant (Heavy outlines show alternatives selected for the reference waste management system.)

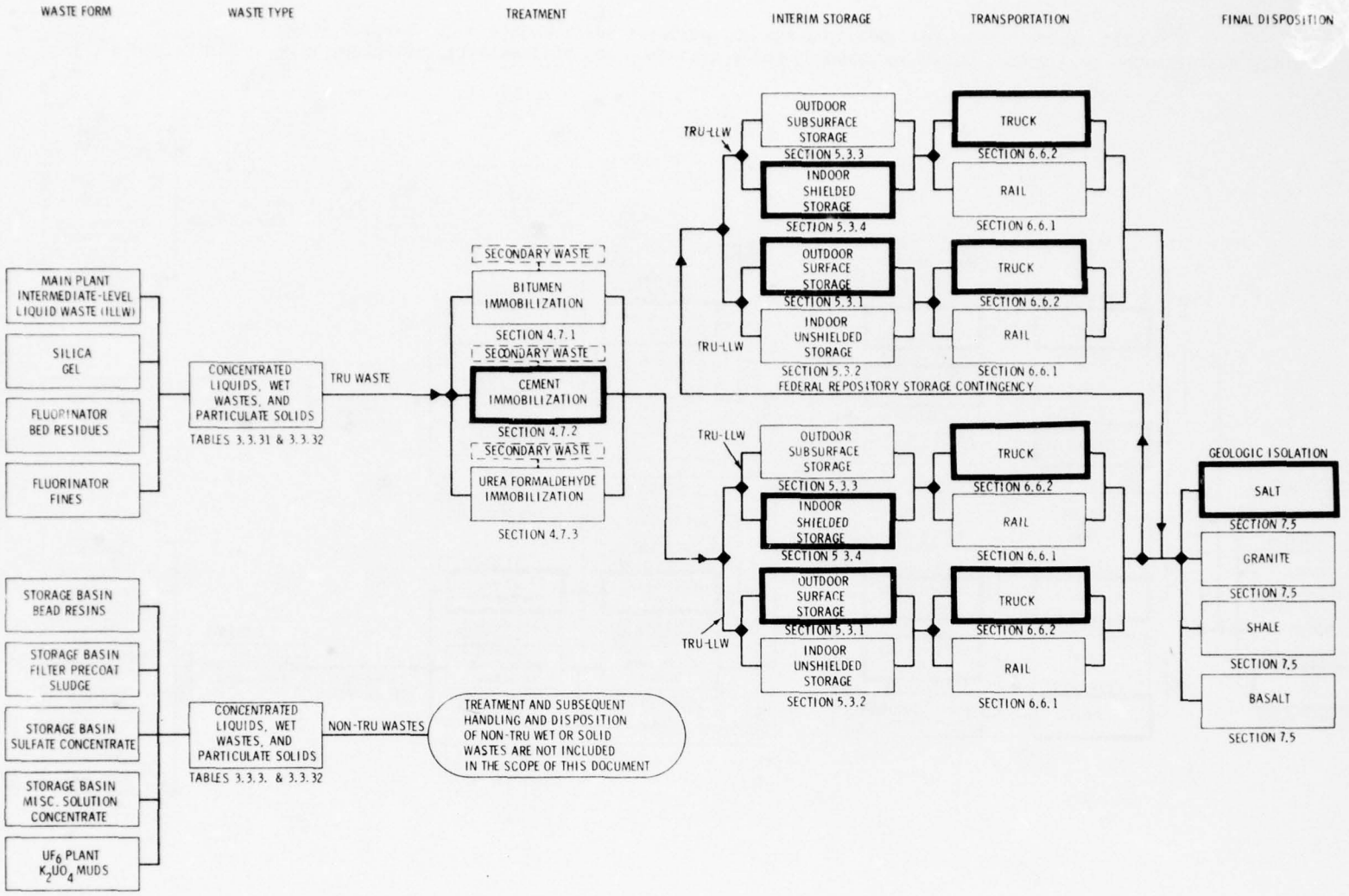

FIGURE 3.4.5. Waste Management Alternatives for Concentrated Liquids, Wet Wastes, and Particulate Solids from a Fuel Reprocessing Plant (Heavy outlines show alternatives selected for the reference waste managment system.)

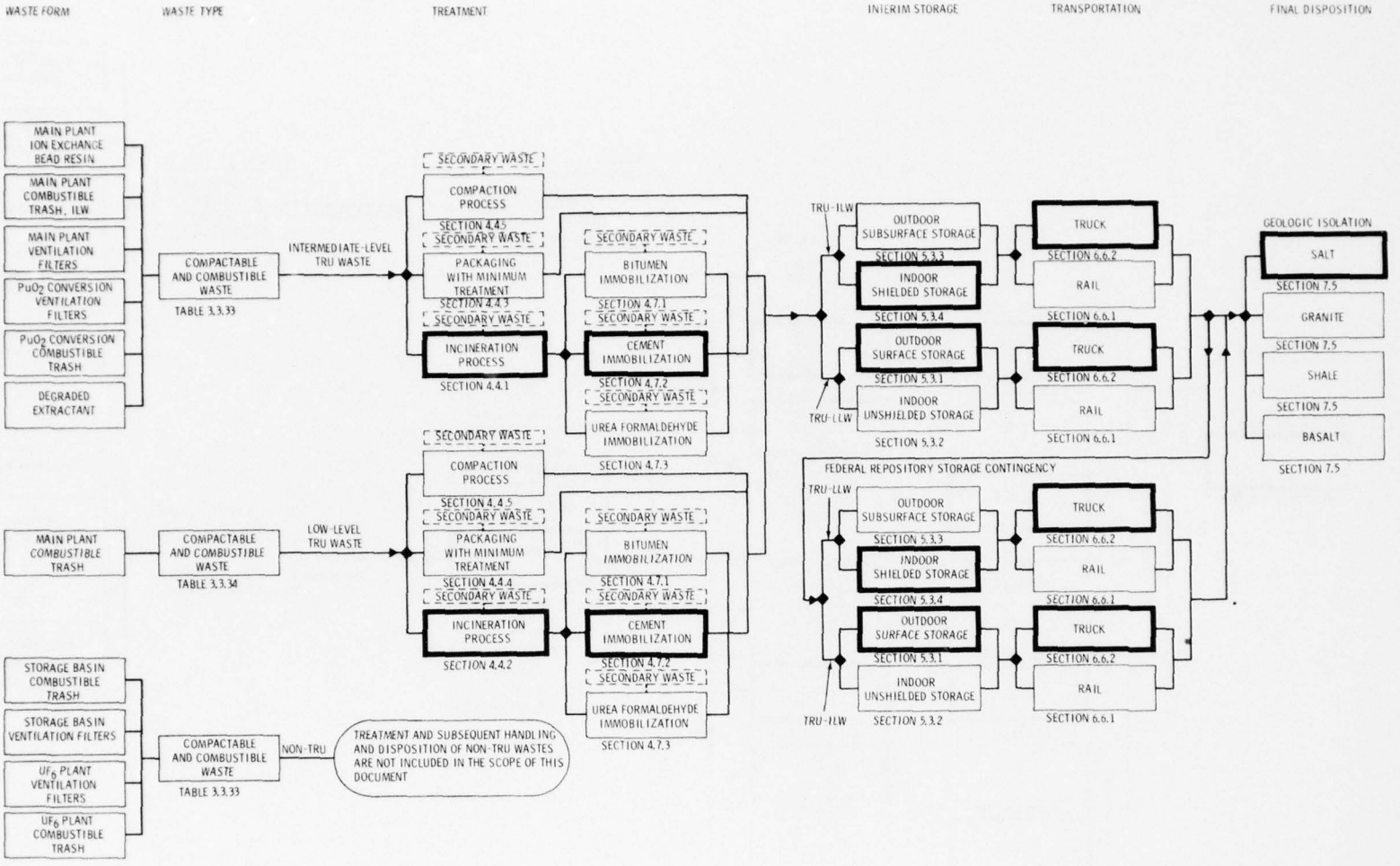

FIGURE 3.4.6. Waste Management Alternatives for Compactable and Combustible Waste from a Fuel Reprocessing Plant (Heavy outlines show alternatives selected for the reference waste management systems.)

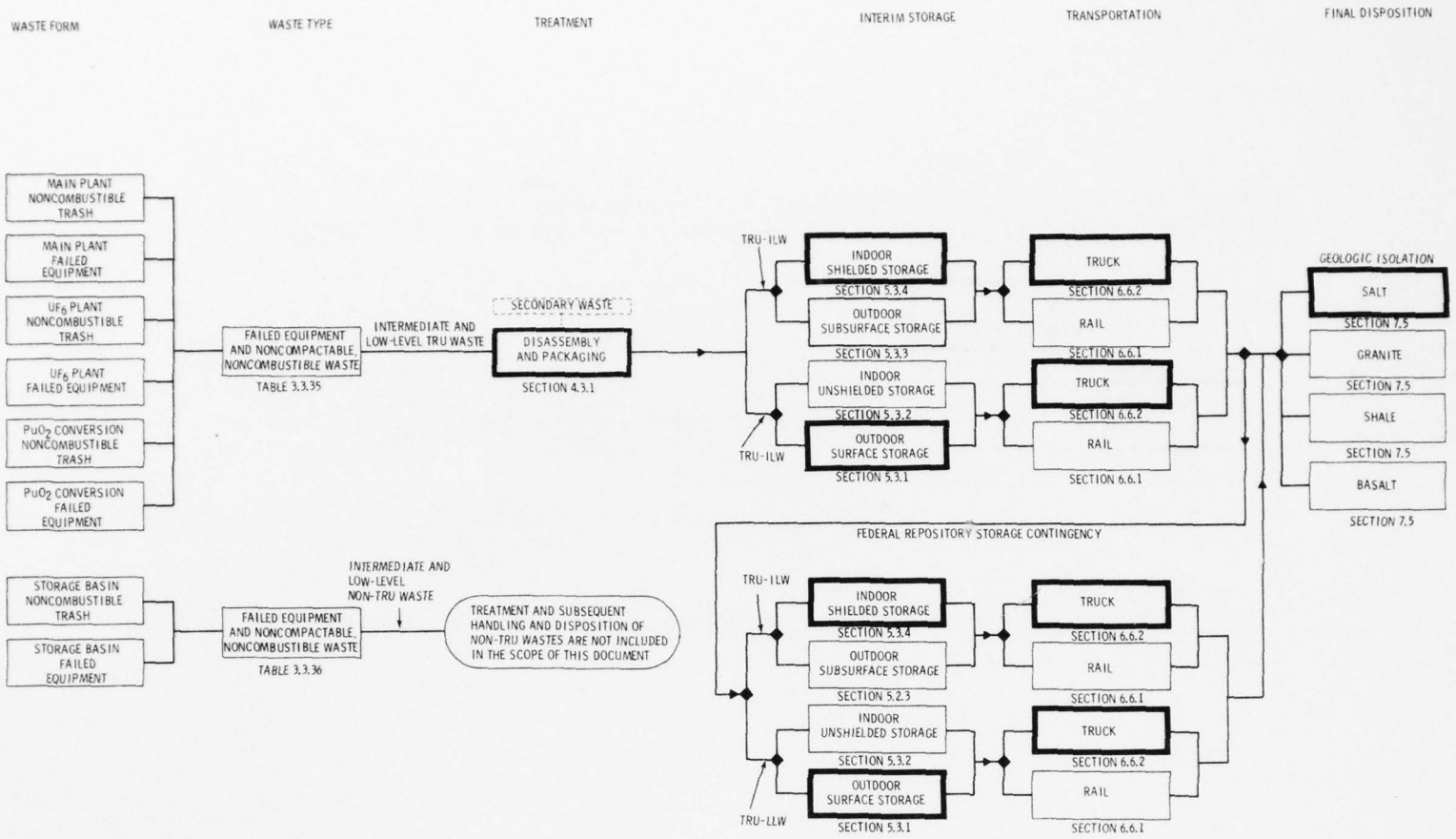

FIGURE 3.4.7. Waste Management Alternatives for Failed Equipment and Noncompactable, Noncombustible Waste from a Fuel Reprocessing Plant (Heavy outlines show alternatives selected for the reference waste management system.)

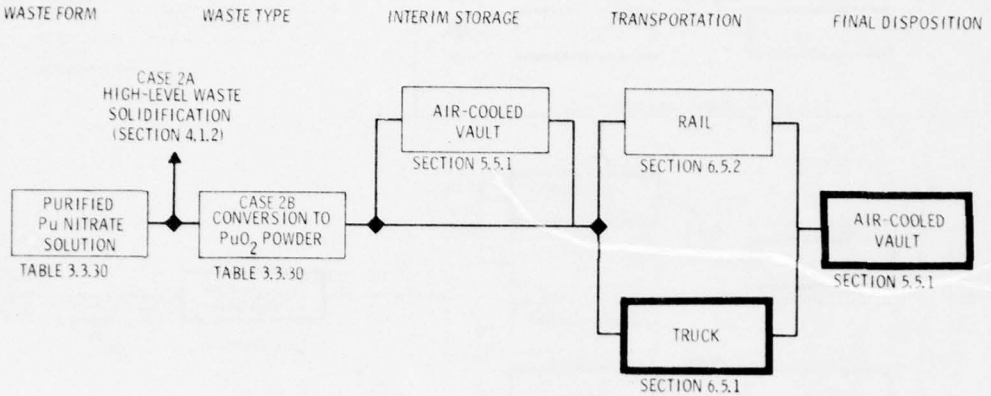

FIGURE 3.4.8. Waste Management Alternatives for Plutonium in Uranium-Only Recycle Fuel Cycle (Heavy outlines show alternatives selected for the reference waste management system.)

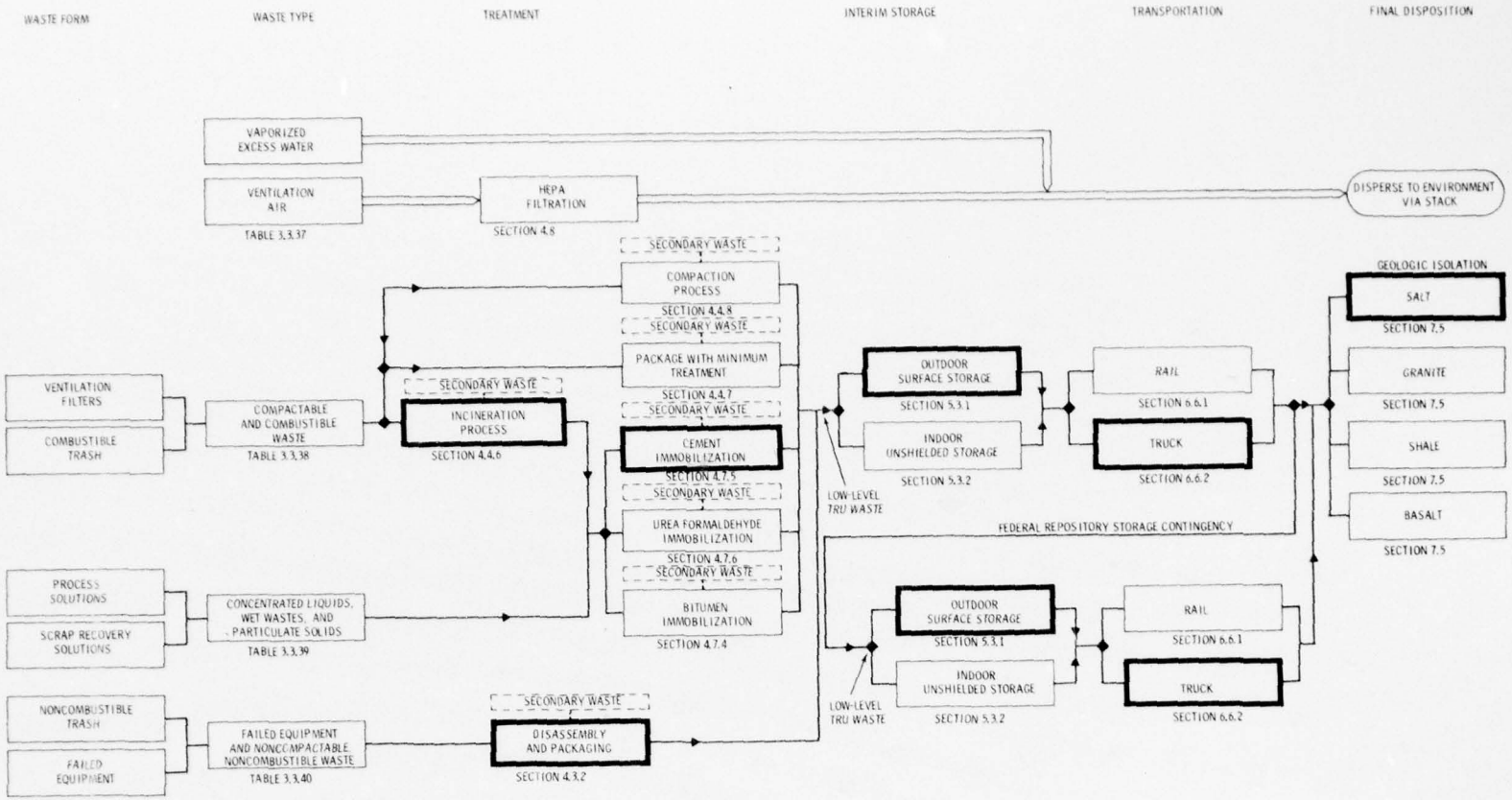

FIGURE 3.4.9. Waste Management Alternatives for a Mixed Oxide Fuel Fabrication Plant (Heavy outlines show alternatives selected for the reference waste management system.)

3.5 SECONDARY WASTES

3.5 SECONDARY WASTES

Secondary wastes are the wastes generated during the treatment, storage, transportation and disposal of primary wastes. They are low- and intermediate-level wastes of relatively small volume that arise solely from management of the primary wastes. An assessment of the technology for radioactive waste management is, therefore, incomplete unless the impact of the secondary wastes is included. This section characterizes the nongaseous* secondary wastes produced by technologies for managing transuranic (TRU) wastes, as outlined in the previous section and as described in detail in Section 4 of this report.

3.5.1 Characterization of Secondary Wastes

Nongaseous secondary wastes arising from operation of the reference waste management system (defined in Section 3.4.4) at the fuel reprocessing plant and mixed oxide fuel fabrication plant are shown in Tables 3.5.1 and 3.5.2, respectively.

The nongaseous secondary wastes are classified into three categories: 1) compactable and combustible waste, 2) concentrated liquids, wet wastes and particulate solids, and 3) failed equipment and noncompactable, noncombustible waste.

3.5.2 Disposition and Treatment of Secondary Wastes

In most instances, the secondary wastes can be recycled, i.e., combined with corresponding primary waste streams for treatment. This is possible because in almost all instances the secondary wastes do not have any distinctive properties that alter the treatment process requirements. An exception involves incineration of primary wastes containing halogenated plastics. The secondary wet wastes resulting from this treatment task contain halogens, which create corrosion problems. Special equipment must be provided for concentrating and neutralizing this secondary waste before it can be combined with primary liquid waste for immobilization.

The overall impact of secondary wastes is an increase in the volume of wastes that must undergo treatment, handling, transportation and disposal. Since the activity introduced into the waste management system is not changed (it depends only on the primary wastes), the specific activity of the treated wastes is actually reduced because of dilution. Evaluation of the effects of secondary wastes is basically, then, an evaluation of the effects of the incremental increase in throughput imposed on the management system.

Different treatment processes produce different volumes of secondary waste and this can become an important aspect in comparing competitive waste treatment options. Tables describing the secondary wastes produced by each waste treatment alternative and each interim storage alternative are included in Sections 4 and 5, respectively.

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* The gaseous secondary wastes for the alternative technologies are described as part of the process descriptions for each technology in Section 4 of this report.

TABLE 3.5.1. Fuel Reprocessing Plant TRU Secondary Wastes for Reference Fuel Cycle[a]

Waste Category	Components, Wt%		Density, kg/m³	Volume, m³/MTHM	Radionuclide Contents[b] Fission Products[c]		Actinides[d]		Activation Products[e]
Noncombustible and Noncompactable Trash	Metal	100	500	0.075	^{3}H	1.5×10^{-5}	U	6×10^{-8}	1×10^{-4}
					^{85}Kr	0	Pu	7×10^{-8}	
					^{129}I	3×10^{-8}	All others	1×10^{-5}	
					All others	1×10^{-5}			
Concentrated Liquids, Wet Wastes and Particulate Solids									
Incinerator Ash	Ash	100	230	0.085	^{3}H, ^{85}Kr	0	U	1×10^{-5}	1×10^{-6}
					^{129}I	2×10^{-6}	Pu	5.8×10^{-4}	
					Zr-Nb, Ru-Rh	1×10^{-5}	All others	2.2×10^{-6}	
					All others	1.5×10^{-6}			
Incinerator Scrub Solution	Solids	25	1200	0.185	^{3}H	1×10^{-7}	Pu	2.5×10^{-5}	1×10^{-9}
	Water	75			^{85}Kr	0	All others	2.5×10^{-7}	
					^{129}I	1×10^{-4}			
					Zr-Nb, Ru-Rh	2.5×10^{-7}			
					All others	2.5×10^{-8}			
Miscellaneous	Solids	25	1200	0.017	^{3}H	3×10^{-5}	U	5×10^{-5}	1×10^{-4}
	Water	75			^{85}Kr	0	Pu	9×10^{-5}	
					^{129}I	3×10^{-6}	All others	6×10^{-7}	
					All others	1×10^{-6}			
Compactable and Combustible Waste	Paper/Rags	40	120	0.28	^{3}H	7×10^{-7}	U	6×10^{-9}	1×10^{-6}
	PVC	20			^{85}Kr	0	Pu	7×10^{-9}	
	Neoprene	16			^{129}I	1×10^{-5}	All others	2×10^{-7}	
	Polyethylene	14			All others	2×10^{-7}			
	Latex	6							
	Wood	4							

a. The reference fuel cycle is defined in Section 3.4.
b. Assuming uranium and plutonium recycle, 2000 MTHM/year reprocessed 1.5 years out-of-reactor.
c. Given as fraction of Ci values listed in Table 3.3.9.
d. Given as fraction of Ci values listed in Table 3.3.15.
e. Given as fraction of Ci values listed in Tables 3.3.6 and 3.3.7.

TABLE 3.5.2. Mixed Oxide Fuel Fabrication Plant Secondary Wastes for Reference Fuel Cycle[a]

Waste Category	Components, Wt%		Density, kg/m³	Volume, m³/MT MOX fuel	Radionuclide Content[b] Actinides[c]	
Noncombustible and Noncompactable Trash	Metal	100	500	0.005	Am	4×10^{-7}
					All others	1×10^{-8}
Concentrated Liquids, Wet Wastes, and Particulate Solids						
Incinerator Ash	Ash	100	240	0.021		3×10^{-4}
Incinerator Scrub Solution	Solids	2	1020	0.32		3×10^{-7}
	Water	98				
Miscellaneous	Solids	25	1200	0.004	Am	2×10^{-5}
	Water	75			All others	3×10^{-5}
Compactable and Combustible Waste	PVC	33	120	0.03	Am	2×10^{-7}
	Collulosics	30			All others	1×10^{-9}
	Polyethylene	18				
	Latex	9				
	Neoprene	9				
	Styrene	1				

a. The reference fuel cycle is defined in Section 3.4.
b. Assuming plutonium in MOX fuel is from fuel reprocessed 1.5 years out-of reator. Fuel is fabricated 1 year later.
c. Given as fraction of Ci values listed in Table 3.3.18.

3.6 SCOPE OF TECHNOLOGY AND FACILITY DESCRIPTIONS

3.6 SCOPE OF TECHNOLOGY AND FACILITY DESCRIPTIONS

To develop the bases for the waste management technology and facility descriptions, experts in 17 areas of waste management technology from twelve DOE contractors and laboratories were appointed as technology characterization Task Leaders. These Task Leaders were assigned responsiblity for selecting the particular alternatives to be evaluated and the reference process or concept for each of the alternatives in their field of expertise and for developing facility descriptions for each waste management alternative. The technology characterization tasks included:

- high-level liquid waste solidification
- fuel residue treatment
- failed equipment and noncombustible waste treatment
- combustible and compactable waste treatment
- non-high-level liquid waste treatment
- concentrated liquids, wet wastes and particulate solids immobilization
- off-gas particulate treatment
- volatile radioisotope treatment
- high-level liquid waste interim storage
- fuel residue interim storage
- non-high-level solid waste interim storage
- solidified HLW interim storage
- plutonium storage
- spent fuel interim storage and spent fuel packaging
- waste transportation
- geologic isolation
- facility decommissioning

In addition to the technology characterization task assignments, six other task leaders were assigned responsibility for system tasks, either providing input for or analyzing information from all of the technology areas. These system tasks included:

- waste characterization
- fuel cycle and systems analysis
- safeguards analysis
- accident analysis
- cost analysis
- criticality and shielding analysis

The laboratories and organizations participating in these task assignments are identified in the Introduction (Section 2.0).

The alternative processes or concepts to be evaluated in each technology area were selected from those identified as commercialized or available technology. A previous ERDA study (ERDA-76-43) was the primary reference. Commercialized technology was defined as technology in routine industrial use; available technology was technology that had been developed sufficiently to permit design and construction of a full-scale commercial installation, though

design verification tests might be required. In the case of geologic repositories, the technology of mine construction is well developed but the technology of site characterization and site selection requires further development.

The facility descriptions developed by the technology characterization task groups provided the following kinds of information:

- technical description of the facility
- chemical and process flowsheets
- descriptions of major process equipment
- descriptions of shielding requirements
- descriptions of operating and maintenance requirements
- estimates of materials and utility requirements
- estimates of staffing requirements
- identification of any hazardous materials required
- general descriptions of required structures
- any special siting requirements
- estimates of any radioactivity or non-radioactive chemicals, particulate or gaseous releases, and any heat releases
- description of any significant odor or noise impacts
- description of potential accidents (see Section 3.7 for a discussion of accident analysis bases).

Because of variations in the amount of facility design information available the scope and content of the task group facility descriptions varied.

An architect engineering (A-E) firm (Bechtel, Inc.) with experience in design and construction of nuclear fuel cycle facilities was contracted to complete the facility descriptions prepared by the tasks groups to an extent sufficient to develop preliminary capital cost estimates. They were also charged with preparing the capital cost estimates and estimating construction requirements and impacts. The descriptions in all cases are preliminary conceptual design descriptions sufficient for preliminary cost estimating purposes. (Development of the cost estimates is discussed in more detail in Section 3.8). A great deal more design effort would be required prior to construction of any such facilities.

In addition to the capital cost estimates, the A-E developed estimates of the following:

- construction schedules
- construction labor force size, composition, and schedule
- site and materials requirements
- transport requirements
- onsite and offsite cost breakdown
- identification of any unusual environmental effects.

3.6.1 Organization and Content of Technology Sections

The information developed by the task groups and the A-E was critically reviewed and consolidated into a consistent format by a coordinating team at the Pacific Northwest Laboratory (PNL), (operated for the Department of Energy by Battelle Memorial Institute). Subsections on accident and cost analyses and on safeguards and physical protection requirements were also developed at PNL based on information supplied by the Task groups and the A-E and supplemented from other sources as necessary for completeness. An intensive effort was made to use and present a consistent data base throughout the report.

In this report the waste management functions are subdivided by major categories in separate sections as follows:

Section 4 Waste Treatment Technology
Section 5 Interim Storage Technology
Section 6 Waste Transportation Technology
Section 7 Final Isolation Technology
Section 8 Retired Facilities Decommissioning Technology

The final two sections in this report (Sections 9 and 10) do not describe waste management technology. Section 9 discusses thorium fuel cycle wastes and Section 10 uses data developed in the preceding sections to develop a complete waste management systems analysis.

The organization of a typical technology section is as follows:

x.x Waste Management Function

x.x.1 Alternative "A"
x.x.2 Alternative "B"
x.x.3 Other alternatives
x.x.4 Physical Protection and Safeguards Requirements.

For example, in Section 4 Waste Treatment Technology, 4.1 High Level Liquid Waste Solidification identifies a waste management function and 4.1.1 High-Level Waste Vitrification and 4.1.2 High-Level Waste Calcination identify the two alternative processes that are described.

Section 4.1.3 contains a brief discussion of other HLLW solidification alternatives not described in detail and Section 4.1.4 contains a discussion of safeguards and physical protection requirements for HLLW solidification processes.

As far as practical the descriptions of each alternative process or concept are presented in a standard twelve component format identified by four-digit subheadings (x.x.x.x). This format is followed closely in Sections 4 and 5, which are the largest technology sections. The format is modified for Sections 6 and 8 because not all of the topics are appropriate. In Section 7 most of the same topics are covered but at a three-digit subheading level.

The twelve sub-headings and the nature of the information found under each topic are as follows:

x.x.x.1 Reference Process or Concept Alternatives

Describes the available process or concept alternatives, identifies the reference process and the reasons for selecting the reference process.

x.x.x.2 Reference Facility Design Bases

Describes reference facility capacity, special requirements or limitations, assumptions, etc.

x.x.x.3 References Process (or Operation)

Describes the reference process or alternatively how the facility operates as in the case of a storage facility.

This section includes a flowsheet of the process or flow diagram of the operations. In the waste treatment sections two important tables are also included here, one listing the quantities of major radionuclides present in the input to the process or facility and one describing the product of the process or facility. By reference to tables in Section 3.3 more detailed radionuclide content data can be obtained. The product description includes such information as the volume, the density, the ratio of the volume after treatment to that before treatment, the number of containers per year within each of several surface dose rate ranges (<0.2 R/hr, 0.2-1 R/hr, 1-10 R/hr, and >10 R/hr), and the radionuclides present in the product (as a fraction of the input to the process or facility). These tables are not required in the waste storage sections. Here the input wastes are described in tables presented in the introduction portion of the sections.

x.x.x.4 Reference Facility Description

A description of the conceptual facility is presented here. Major components, systems, and equipment items are identified and described. Plan and elevation view drawings of the facility are included here. Special considerations such as shielding requirements are also addressed here. It should be kept in mind that not all of the construction details will be found on these preliminary-design drawings. For example, details such as viewing windows, entry ways, tie-ins with other facilities, etc. are not necessarily shown. However, allowances are made for these items in the cost estimates.

x.x.x.5 Reference Facility Operating and Maintenance Requirements

Typical items covered here include the mode of operation (around-the-clock, one-shift per day, intermittent, etc.), whether or not operation and maintenance activities need to be performed remotely, special replacement requirements, hazardous materials usage, staffing requirements, and supply and utilities requirements. The "water usage" values listed under utilities requirements refer only to that amount of water that is consumed, i.e., discharged as vapor from a cooling tower or otherwise made unavailable for unrestricted release in the liquid form and ultimate return to its source.

x.x.x.6 Reference Facility Secondary Waste

Secondary wastes resulting from operation of this facility are described here (see Section 3.5 for further details).

x.x.x.7 Reference Facility Emissions

The emissions that result from operation of the "reference facility" are described here. Important emissions of non-radiologic as well as radiologic significance are listed. The annual quantities of the non-radiologic materials are tabulated. The radionuclide emissions are listed as a fraction of the input to the facility so that the effectiveness of the containment practices will be readily apparent. Detailed isotopic releases can be obtained by reference to the input tables. In many cases the radionuclide-bearing emission from the "reference facility" is routed to a point in the primary facility where it receives additional treatment before it is discharged to the atmosphere; the fractional releases tabulated in these subsections include the effects of these additional treatments where applicable.

Also included as an emission is an estimate of the integrated annual release from minor accidents. This estimate was developed by weighing the minor accident releases by their expected frequencies and summing the quantities for all identified minor accidents. In addition, a contingency was included to account for unidentified minor accidents and to compensate for the uncertainty in expected frequency information.

Other listed emissions include those from a cooling tower (evaporation, drift, and blowdown). To improve standardization between sections, a "reference cooling tower" was developed and used to calculate these emissions from the indicated releases of heat to a cooling-water system. The "reference cooling tower" releases water as evaporation and drift at a temperature of 38°C and as blowdown at 27°C. The releases (per MW of heat duty) are: 414 g H_2O/sec as evaporation, 2 g H_2O/sec as drift, and 72 g H_2O/sec as blowdown. The water make-up flow is thus 488 g H_2O/sec-MW.

x.x.x.8 Reference Facility Decommissioning Considerations

Features of importance here include the estimated useful life, the level of residual activity, and design features to simplify decommissioning.

x.x.x.9 Reference Facility Postulated Accidents

Scenarios and radioactivity release estimates for postulated accidents are described here (see Section 3.7 for additional details).

x.x.x.10 Reference Facility Costs

Capital, operating and levelized unit costs are described here (see Section 3.8 for additonal details).

x.x.x.11 Reference Facility Construction Requirements

Project schedules, manpower requirements, site requirements, water requirements, construction material requirements and energy (i.e., fuel) requirements are detailed here. Construction schedules assume no untoward delays. Licensing difficulties could significantly extend the estimated project schedules.

x.x.x.12 Effects of Fuel Cycle Options

Brief discussions of how fuel cycles other than the one used for design basis would affect the "reference facility" are presented here.

3.6.2 Reference Site

Since this was a generic study, a number of assumptions were necessary to establish a basis for design and construction requirements. The following generic site characteristics and construction conditions and requirements were assumed for all facilities unless otherwise defined in the individual facility descriptions:

3.6.2.1 Reference Site Location

The site is assumed to be located in the midwestern United States. The nearest town with a population of at least 2,000 people is 12.9 km (8 miles) from the site. Nearby large cities having populations of 40,000 and 2,000,000 are 32.2 and 48.3 km (20 and 30 miles) from the site, respectively.

3.6.2.2 Meteorological Conditions at the Reference Site

Wind.

Prevailing directions	Northwest and South through Southeast
Maximum velocity	161 km/hr (100 mph)
Average velocity	0-16.1 km/hr (0-10 mph)
Design velocity (basic wind speed)	144.8 km/hr (90 mph)
Design pressure	Per ANSI 58.1 with basic wind speed of 90 mph

Tornado. For Category I construction, the design tornado is the Region 1, Design Basis Tornado, as specified in NRC Regulatory Guide 1.76.

Tornado Missiles. For Category I construction, design missiles are in accordance with ASCE Structural Design of Nuclear Power Plants, December 8-10, 1975, Volume 1A, as specified by the engineer and approved by the NRC.

Precipitation.

Annual average precipitation	77 cm/yr (30 in.yr)
Maximum precipitation	13 cm/day (5 in./day)
Design maximum rate, peak 1 hr rate 50 yr recurrence	6.4 cm/hr (2.5 in./hr)
Design maximum duration 10.16 cm (4 in.) total precipitation 50 yr recurrence	6 hr

Snow. The design load is 150 kg/m^2 (30 lbs/ft^2)

Temperature.

Summer maximum (July)	43°C (100°F)
inter minimum (January)	-34°C (-30°F)
Design maximum, summer	
dry bulb	32°C (90°F)
wet bulb	-24°C (75°F)
Design minimum, winter	-26°C (-15°F)

3.6.2.3 Surface Conditions and Transportation Availability at the Reference Site

Obstructions. No major obstructions to facility construction exist.

Topography. Ground surface is essentially even and moderately sloping, providing good drainage without erosion.

Vegetation. Ground cover is moderately low and has no extensive tree growth. Some grubbing and clearing for construction of facilities is required.

Drainage. Drainage is good and without swamps. Runoff is to natural streams, drainage channels, or basins.

Flooding. Elevation of the site is such that with a design storm occurrence, no flooding of facilities will occur either from local runoff or from flooding of any nearby streams or lakes. Location of water reservoirs and dams is such that no flooding will occur in the unlikely event of a dam failure due to earthquake or other causes.

3.6.2.4 Transportation and Utility Availability at the Reference Site

Roads. Approximately 1.6 km (1 mile) of new road construction is required to provide access to the site from an existing U.S. highway.

Railroads. Approximately 3.2 km (2 miles) of new railroad is required to provide a rail spur service to this site.

Utilities. At the start of construction, water supply, electrical power, fuel oil, steam, and compressed air will not be available at the site. To facilitate construction activities, the initial site preparation includes installation of access road, rail spur, water supply wells, commercial power service, steam boilers, air compressors, and fuel oil storage. Until these plant facilities are installed, construction water will be imported, and compressed air and commercial power will be provided by portable units.

3.6.2.5 Subsurface Conditions at the Reference Site

Obstructions. No major underground obstructions to facility construction exist.

Soils. A thin layer of glacial till soil, generally less than 3 m (10 ft) thick, covers the underlying rock.

Rock. The soil cover is underlain by sandstone and shale rock formations which can support loads up to 73,000 kg/m^2 (15,000 lbs/ft^2).

Groundwater. Groundwater exists relatively near the surface and will require special dewatering for foundations and deep excavations. Below grade concrete walls and slabs of rooms, cell vaults, basins, etc., will require waterproofing.

Frost. Design ground penetration is 1 m (40 in.).

Cavities. No cavities or small voids exist in the soils or rock underlying the site. No calcareous materials or other chemicals are present which might form future solution cavities.

3.6.2.6 Geologic and Seismic Conditions at the Reference Site

Faults. There is no indication that faulting affects the site area. The nearest known or inferred fault is 37 km (23 miles) from the site and is considered not capable.

Structure. Other than Seismic Design Category 1, structures are designed to resist earthquake effects based on UBC Zone 2. Category 1 structures, on the other hand, are designed to resist earthquake effects based on the maximum ground accelerations listed below with amplification factors from NRC Regulatory Guide 1.60, and damping values from NRC Regulatory Guide 1.61.

Operating Basis Earthquake (OBE)	0.1 g
Safe Shutdown Earthquake (SSE)	0.2 g

Liquefaction Potential. Soils and groundwater conditions are such that when subjected to seismicity, liquefaction of the soils will not occur.

3.7 BASIS FOR ACCIDENT ANALYSIS

3.7 BASIS FOR ACCIDENT ANALYSIS

The accidents considered in this report have been experienced or can be realistically postulated for licensed nuclear waste processing and storage facilities. However, no attempt was made to exhaustively analyze all conceivable accidents. Though radioactive material inventories at nuclear waste processing facilities represent an inherent potential hazard regardless of the treatment or storage alternatives selected, accident types and corresponding sequences of events are characteristic of the specific process alternative and equipment. Thus, the potential accident spectrum might be altered for specific licensed waste management systems. Many of the postulated accidents might be eliminated by advanced plant design or operational techniques. Recognizing these limitations, the accident spectrum described in this report is believed to be generically representative of potential accidents in commercial radioactive facilities.

The main emphasis of accident analyses in this report is to identify accidents with potential for offsite releases. In addition, it is realized that some accidents may also expose plant workers. Several accidents with potential for internal contamination of workers are identified in the individual accident source terms. Accidents which may increase radiation levels in occupied zones and could result in increased worker exposures are identified by footnotes in the list of accidents in Appendix 3A. Generally these accidents have a significant potential for offsite releases or disruption of plant activities.

Before a construction permit could be issued for facilities such as those described in this report, more exhaustive analyses than those developed for this study would be carried out during licensing procedures under Federal Regulation 10 CFR 50 or equivalent regulations expected to be adopted for geologic repositories. 10 CFR 50 requires an environmental impact statement and a preliminary safety analysis report based on preliminary design details.[1] In addition, the facility operating license requires a final safety analysis report based on final design details. These safety analysis reports must define the facility safety hazards and describe the safety systems that will keep accidental releases within regulatory limits.

The prime objective of the analyses in this report is an estimate of potential hazardous material releases (called source terms). These provide the bases for estimates of potential exposures to employees and nearby populations. These exposure estimates are presented in DOE/ET-0029. A complete source term includes information such as type and quantity of nuclides, chemical form, release point, rate of release, and quantity of diluent (air or water) accompanying the release. However, the point of release and diluent flow rates are the same for most accidents at a particular facility; for example, the main stack and stack effluent at a fuel reprocessing plant (FRP). In these cases the necessary source term data reduces to released material quantities and the rate of release.

An umbrella source term concept is used to limit the number of accidents requiring detailed impact analysis in DOE/ET-0029. Viewed independently of accident initiation sequences and frequencies, the design-basis accident source terms can be grouped by release severity for environmental consequence analyses. Forty-one different release groups are defined based on similar release pathways, chemical form, accident severity category, and isotope types released

(fission products, activation products, and actinides). The largest release from any of the accidents in a group of accidents was selected as the umbrella source term for that group.

The first step in accident source term definition is identification of representative accidents. Each waste treatment and storage technology was examined by a technology task group to postulate potential accidents which might result in offsite releases or significant impact on plant operations. The accident lists were then reviewed by safety specialists to assure a consistent and comprehensive analysis. This is a speculative process, especially for those facilities that are conceptual and thus have no operating experience. In these cases, the depth of the accident analysis is limited by the depth of the conceptual design details available for the facilities.

Source terms were developed using successive release fractions. These fractions can be defined as the amount of radioactivity released in an accident. Specifically, the release fraction is the fraction of radionuclide inventory that is released to the next containment barrier or to the environment.

The radioactivity released in an accident may be substantially reduced by one or more barriers, such as a high-efficiency particulate air (HEPA) filter bank. The radioactivity released to the environment is estimated by multiplying the product of the release fraction for each release mechanism and containment barrier (e.g., the accident, process equipment, HEPA filters, etc.) by the radionuclide inventories involved in the operation. Where multiple waste treatment options are examined, analysis is based on the reference system (see Section 3.4) waste form.

Another important factor for assessing the impact of potential accidents is accident frequency. Accident frequency estimates were developed for identified accidents where possible. Many of these estimates are based on previous experience with similar equipment, but many others are engineering judgment based on review of the conceptual designs. The given frequency values should be considered order of magnitude estimates. A listed frequency for a specific event is believed to be an upper limit estimate; however, they have a substantial uncertainty since accident frequencies are sensitive to the type, manufacture, installation, and operation of the equipment.

Following source term and frequency definition, the lists of representative accident scenarios were classified into three accident severity groups:

- Minor: relatively frequent occurrences involving process interruptions without potential for significant release of radioactive or other hazardous materials.
- Moderate: infrequent events with potential for small material release, major equipment damage, or the creation of radiation fields in occupied zones which could result in occupational exposures exceeding 10 CFR 20 limits (5 rem/yr).
- Severe: unlikely events with a potential for significant radiation hazards; these events are postulated to establish performance requirements for plant safety systems. Accidental releases of sufficient radionuclides to cause occupational exposures which could result in detectable clinical effects (assumed to be >25 rem) are included in this

category. Since the safety systems are designed to confine and mitigate the impact of these severe accidents, the actual release may be no larger than the "minor" or "moderate" class accidents.

The three accident classifications cover the spectrum of design-basis accidents. Nondesign-basis accidents include all accidents which exceed site criteria* or involve concurrent independent failure of process and multiple containment system barriers. By virtue of plant design and operational techniques, the possibility of a nondesign-basis accident is extremely unlikely during the design life of the waste treatment or storage facility. However, because of the long period of required containment, several nondesign-basis accidents for geologic isolation of nuclear waste are postulated. Nondesign-basis accidents for other facilities, which all have expected useful lives of less than 100 years, are not considered in this report.

Releases of radioactive material to the environment result from two different events: 1) normal operational releases and 2) releases resulting from accidents. Operational releases result from routine handling or processing of radioactive materials and are limited by the containment system design and performance. They are expected to occur at a relatively uniform rate over the life of the plant. Accidental releases occur intermittently because of operational errors or system component and containment failures, and releases are generally inversely proportional to their frequency. The small-release, moderate-frequency minor accidents are characterized in this report in two ways: 1) as a short term intermittent release from a single accident and 2) as an integrated annual release from all minor accidents averaged over one year and added to the normal operational emissions. Because of their low frequency, releases from moderate and severe accidents are characterized only as intermittent releases from individual accidents.

Accidents postulated for this report are presented in each waste management facility description. Following introductory material specific to the analysis and history of accidents for each technology, accident descriptions are reported by severity category in the following four column tabular format. This format concisely presents large amounts of information describing the sequential nature of accident propagation and the way plant safety systems intervene to mitigate potential accidental releases.

- Column 1 - accident number: a three-digit number that identifies each accident by technology, descriptive title, and estimated frequency of occurrence
- Column 2 - sequence-of-events information that shows the dynamic path of the accident
- Column 3 - safety systems that explain how an accident can be prevented by design, discovered by alarms, and/or mitigated by a confinement system and/or operator action
- Column 4 - amount of release, the expected source term, i.e., the material specification, location, and duration of the expected potential release.

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* Site criteria include 1) definition of the maximum credible earthquake, surface faulting, floods, and wind velocities based on historical evidence, local and regional geology, and expert judgment; 2) local and regional demography; and 3) proximity and definition of hazards caused by man.

A listing of all postulated accidents and a cross reference listing under the appropriate umbrella source term is provided in Appendix 3A. Accidents with a potential for external worker exposure are also identified. Umbrella source terms and the results of the consequence analysis are presented in DOE/ET/0029.

The bases for the nuclear criticality calculations carried out for this study are described in Appendix 3B.

REFERENCES FOR SECTION 3.7

1. Title 10, Code of Federal Regulations, Part 50 (10 CFR 50).

3.8 COST ANALYSIS BASES

3.8 COST ANALYSIS BASES

One of the important aspects of radioactive waste management is cost. A great deal of concern has been expressed regarding possible costs for managing the radioactive wastes produced in the commercial nuclear fuel cycle. To help place this issue in perspective and to provide a basis for comparing waste management costs for alternative fuel cycles, subsequent sections of this report include estimates of capital, operating, and unit costs for the various waste management facilities described. This section discusses the assumptions and methodology used to derive these cost estimates, as well as the bases for determining estimate uncertainty ranges.

A constant dollar method of analysis is employed in which all costs, both present and future, are expressed in terms of the buying power of the dollar in mid-1976. This is not meant to imply that inflation will not occur; rather, cost relationships can be more easily understood and placed in perspective if they are stated in constant dollar terms. It is reasonable to assume that, over the long term, the estimated costs developed in this study will increase at a rate comparable to the general rate of inflation.

In general, capital costs are derived by estimating requirements for major equipment, buildings and structures, site improvements, and construction labor. These direct cost estimates are then factored to generate other direct costs, indirect costs, architect-engineer costs, and owner's costs. Operating costs include all cost items identified with the labor force or production. The number of man-hours, quantities of materials, and requirements for utilities have been derived in each case from the reference facility descriptions. The allowances for maintenance, overhead, and miscellaneous costs are derived by factoring either capital or direct labor costs. Levelized unit costs are calculated charges per unit of production sufficient to earn a specified return on outstanding capital investment; pay all operating expenses, including taxes and insurance; and, over the life of the project, recover the capital investment.

The capital cost methodology outlined in this section is used to estimate all of the capital costs given in this report except for those of the transportation facilities, which are discussed in Section 6. The operating cost methodology applies to all waste management functions except transportation and geologic repositories; cost development for these two functions is described in Sections 6 and 7, respectively. The levelized unit cost discussion gives the data used in estimating the cost of money and other calculational parameters; it also addresses the effect of economic plant life on unit costs.

3.8.1 Bases for Capital Cost Estimates

The capital cost estimates in this report are presented using terms and phrases whose meaning may not be self-evident. Terms not defined elsewhere in this text are defined below:

Allowance item:	A number, arrived at by judgment, that represents material or equipment cost that cannot be developed otherwise because of the absence of design detail.

Contingency:	The amount of money added to the estimated cost of a project to cover certain areas of cost uncertainty and reduce the probability of understating the project cost estimate. With the contingency added, there is a more nearly equal probability of a cost underrun or overrun.
Cost of money:	Weighted cost of debt and equity financing. "Cost of money" is used synonymously with "cost of capital."
Scope:	Refers to the work within the purview of the architect-engineer, the contractor, and the owner (e.g., construction, engineering, procurement, and management). Scope also refers to the description and definition of the processes, design details, capacities, physical sizes, and so on required for developing the cost estimate.
Site-related:	Refers to facilities whose design are unique to a given site. The term "site-sensitive" is also used in this sense.
Start-up:	The testing of equipment and systems, under actual or simulated operating conditions, that takes place during the period between mechanical completion and commercial operation of a given facility to assure that the facility is functioning properly in all respects. This activity is included in owner's costs.
Take-off:	The analysis of engineering drawings or sketches to determine the physical quantities of materials required for a facility.

3.8.1.1 Construction Conditions

As a basis for estimating capital cost, the following construction conditions are assumed to prevail at all sites:

- Construction labor follows a 40-hour, single-shift work week schedule. Exceptions include casual overtime (e.g., to complete a concrete pour) and instances where two- or three-shift concrete work operations are planned to meet the construction schedule.
- Severe work stoppages, such as extensive jurisdictional disputes between labor crafts, do not occur during construction.
- The labor force in each craft is adequate; labor need not be imported except for general foremen.
- Craft labor wage rates, including fringe benefits, are those prevailing for a Midwest construction site in mid-1976.

3.8.1.2 Field Costs

The total field costs are the sum of the direct and indirect site construction costs. Descriptions and methodology for pricing the various elements that constitute the field costs are outlined below.

Major Equipment. Major equipment includes all purchased equipment that does not require or involve third-party labor at the construction site for installation or erections. Equipment types include pressure vessels, heat exchangers, cooling towers, tanks, boilers, pumps and drivers, vacuum equipment, compressors and drivers, materials handling equipment, and slave manipulators.

Major equipment costs are derived from estimated prices of similar equipment as indicated in cost estimates developed for other fuel reprocessing plants, radioactive waste disposal facilities, and other plants dealing with the nuclear fuel cycle. Necessary adjustments for capacity, pressure, temperature, complexity, and metallurgy were made; the adjusted prices were then escalated to mid-1976 dollars. In cases where specific design information was limited, cost allowances were developed based on engineering judgment and recent nuclear experience.

Buildings and Structures. For cost estimate purposes, the buildings and structures category comprises all materials, equipment, and labor required to construct the architectural and structural components of a particular facility, including excavation and backfill; painting and coatings; heating, ventilating and air conditioning service; and normal building utilities extending to a point 5-ft outside the building or structure line. Major process equipment and associated piping, electrical, instrumentation, and site improvements are not included in this cost category.

For heavy, concrete-walled structures, costs are based on take-offs from concrete and steel estimates, as well as other building material estimates. For lighter structures, such as administration buildings, costs are estimated by applying unit volume or area cost factors for construction to the estimated volume (area) of the facility.

Bulk Materials. Bulk materials, including instruments and controls, pipe, valves, pipe fittings, pipe coverings, conduit, wire, cable, and cable trays, are materials associated with major process equipment and purchased in quantity for fabrication or assembly and installation at the job site. Generally interpreted, this means all materials that are not defined as major equipment, buildings and structures, and site improvements.

Except for instances where enough information exists to warrant quantity assessments and unit pricing of certain specifically identified material, bulk materials costs are determined either as a function of major equipment costs or as a cost allowance. Factored estimates take recent experience into account and incorporate escalation to mid-1976 dollars.

Site Improvements. Site improvements comprise work within the limits of plant property that is not identified with major equipment, buildings and structures, bulk construction materials, or temporary construction facilities. Examples of the kinds of work in this category include site clearing and grubbing, site grading, roads, railroads, walks, fences, area

lighting, paving, sewers and fire water system (external to buildings), reservoirs and ponds, and landscaping. Site improvement costs are generally determined as a cost allowance based on experience and the plot plan descriptions.

Direct Labor. Direct labor includes all manual and nonmanual personnel that are directly hired by the contractor for performing the work at the project site. Direct labor costs are evaluated from estimated man-hours for erection and installation sequences and operations and craft wage rates and fringe benefits in effect at mid-1976. Labor man-hours are representative of the craft production rates in the area of reference jobsites.

Indirect Site Construction Costs. Indirect site construction costs include material and labor costs that cannot be identified with or allocated to specific direct cost operations in the construction of either the entire project or individual facilities. Examples are costs for contractor's supervision, construction tools, equipment and consumables, temporary construction facilities, utilities, material handling, and cleanup. The contractor's fee is also included in this category. These costs are combined and expressed as a factor of direct labor in each facility estimate.

3.8.1.3 Architect-Engineer Services

The architect-engineer (A-E) manages the project. He also engineers, designs, and procures all plant systems, buildings and structures, and site improvements as defined in the facility descriptions. The cost of A-E services are estimated as a percentage of the total field cost and include overhead costs and fee.

Specific A-E responsibilities include: 1) project management, comprising construction management; project planning, scheduling, and supervision; developing a project cost estimate, a project budget, and project control methods; and implementing project forecasting procedures; 2) engineering and drafting-services, comprising design analysis and the preparation of equipment specifications, design drawings, and engineering information for the Environmental Report (ER), and the Preliminary and Final Safety Analysis Reports (PSAR and FSAR)*; 3) inspection and testing for mechanical integrity prior to start-up; and 4) procurement of equipment and materials for the plant and site facilities, comprising soliciting bids, purchasing equipment and material, inspecting, implementing quality assurance procedures, expediting, and shipping equipment to the jobsite.

3.8.1.4 Owner's Costs

Owner's costs include all costs incurred by the owner within his own organization in connection with the facility construction. These costs normally do not form part of the A-E or contractor scope of work. The various owner's cost components are given in Table 3.8.1. Owner's costs are estimated as a percentage of the total facility construction cost, ranging from 20% to 45%.

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*The ER, PSAR and FSAR are licensing requirements.

TABLE 3.8.1. Components of Owner's Cost

Component	Approximate Percent of Total Owner's Cost
Land acquisition	6
Spare equipment	2
Permits and licenses	4
Interest during construction	60
Start-up	3
Initial inventories	3
Owner's staff	8
Insurance	4
Other	10
	100

3.8.1.5 Costs Not Included

Exclusions from the capital cost estimate are generally limited to the following cost classifications:

- escalation of costs beyond mid-1976 dollars
- process and patent royalties
- general research and development costs
- costs incurred beyond those that reflect the current degree of involvement in securing approvals from regulatory agencies monitoring environmental and safety considerations
- costs generated directly by any governing or regulatory agency for administration, engineering, procurement and construction
- sales/use tax
- nuclear hazards insurance that may be required if nuclear hazards exist on site before completion of project
- housing for construction workers.

3.8.2 Bases for Operating Cost Estimates

Operating costs include labor, process materials, utilities, maintenance, overhead, and other miscellaneous items identified with the labor force or with production. The number of man-hours, the quantities of materials, and the requirements for utilities have been derived in each case from the facility descriptions. Allowances for maintenance, overhead, and miscellaneous costs have been derived as factors either of capital cost or direct labor costs.

3.8.2.1 Direct Labor Costs

Table 3.8.2 shows the costs for direct labor by job classification. This information is used as a basis for operating cost estimates in this study.

TABLE 3.8.2 Salary Schedule for Direct Labor Personnel[1] (1976)

Operations	Salaries, $/yr
Operations supervisor	25,400
Shift supervisor	20,500
Senior control room or chief operator	16,200
Process or power operator	15,000
Equipment operator	14,600
Equipment attendant	14,200
Maintenance	
Maintenance supervisor	27,300
Foreman	18,700
Craftsman	15,800
Helpers	13,500

These costs are based on 1976 labor costs at Hanford, Washington; Washington labor costs are similar to those in the eastern Midwest. Labor costs per man-hour were based on 2000 man-hours per man-year.

Personnel requirements shown in the individual facility descriptions in this report are expressed only in terms of operators, maintenance personnel, and radiation monitors. For facilities requiring a labor force greater than 15-20 persons, direct labor cost estimates were made by allocating personnel to the above categories. For smaller facilities, operators were defined as process or power operators and maintenance personnel as maintenance craftsmen. Radiation monitors were defined as having the same pay scale as equipment operators.

For those facilities in which maintenance must be performed remotely, an additional allowance was made for maintenance labor. Experience at Hanford has been that maintenance labor costs are 2-20 times as much for remote maintenance as for contact maintenance. For this study, maintenance labor costs for remotely maintained facilities were increased by a factor of 5 over contact maintenance costs.

3.8.2.2 Process Materials Costs

Costs for process materials are based on current (1976-77) price quotations for the Hanford operations. Prices quoted include appropriate discounts when bulk quantities are indicated. Prices for consumable hardware (e.g., canisters and storage baskets) were developed by the architect-engineer who prepared the capital cost estimates.

3.8.2.3 Utilities Costs

Cost estimates for utilities are based on the unit costs shown in Table 3.8.3. Cost estimates for facilities within a primary reference plant (i.e., spent fuel storage facility, fuel reprocessing plant, or mixed oxide fuel fabrication plant) only include costs for resources actually used in processing. Normal lighting and heating are assumed to be included in the primary plant's facilities.

TABLE 3.8.3. Utility Costs[2]

Resource	Cost
Electricity	$0.02/kWh
Natural gas	$5.3/100 m^3 ($1.50/1000 ft^3)
Process steam	$3.30/1000 kg ($1.50/1000 lb)
Fuel oil	$0.10/ℓ ($0.40/gal)

3.8.2.4 Maintenance Materials Costs

Maintenance costs for an operating plant are calculated as annual percentages of the initial capital investment. These percentages vary from 2%-5% per year, depending on the amount of equipment and the complexity of operations.[3] Table 3.8.4 illustrates the general criteria used.

TABLE 3.8.4. Basis for Determining Cost of Maintenance Materials

Maintenance Operation	Annual Percentage of Initial Capital Investment
Normal storage operations	2%
Normal processing operations	3%
Complex processing operations with contact maintenance	4%
Complex operations with remote maintenance	5%

3.8.2.5 Overhead Costs

Overhead costs are defined to include:

- salaries of executives, managers, supervisors not directly involved in operations, secretaries and clerical personnel, grounds and building support personnel, guards and technical staff
- employee benefits; office supplies; medical, engineering, employment and other services
- miscellaneous costs not directly related to production
- cost of maintaining offices and buildings.

The first three items were estimated as 110% of direct labor costs; the last item was estimated as 60% of maintenance materials.[4]

3.8.2.6 Miscellaneous Costs

Miscellaneous costs mainly comprise miscellaneous supplies and equipment, outside support services, and any direct costs not otherwise categorized. These costs were estimated at 0.3% per year of initial capital expenditures.[5] Unusual items of significant expense falling in this category were costed separately and added to the estimate.

3.8.3 Working Capital

Working capital is defined as the cash required to operate a facility, i.e., the difference between current assets and current liabilities. This cash is treated as an outflow of funds during the first year of plant operation and as an inflow during the last year of operation. For this report, working capital requirements are estimated at 50% of the first year's operating cost.

3.8.4 Decommissioning Costs

The cost of waste management is defined in this report to include the cost of facility decommissioning. Decommissioning costs for the reference primary facilities--the nuclear power plant, the spent fuel storage facility, the mixed oxide fuel fabrication plant, and the fuel reprocessing plant, including their respective waste management facilities--are developed and shown in Section 8 of this study. The costs to decommission all other facilities not included in decommissioning the primary facilities were estimated at 10% of their capital costs and are incorporated in the levelized unit cost calculations for these waste management facilities. Based on the ratio of decommissioning costs to facility capital for the primary facilities (see Section 8.0), the 10% cost allowance probably overestimates the decommissioning charges. The net effect of the decommissioning charges on levelized unit costs is small, however, because of the cost-of-money discounting effects due to the time periods that elapse before the costs are incurred.

3.8.5 Bases for Levelized Unit Cost Estimates

As the name implies, levelized unit cost estimates are capital and operating charges translated into equivalent, constant (or level) annual unit costs. The unit cost is sufficient to pay any interest charges on debt; pay all operating expenses, taxes and insurance; earn a specified return on outstanding capital; and, over the life of the project, recover the capital investment. For this study a computer program named UNICOST was used to perform these calculations.

Since the calculated unit costs are a function of taxes and returns on equity and debt, it is important to define the assumed ownership of the facilities. Table 3.8.5 gives the relevant ownership information.

The following throughput assumption was made for processing facilities: The facility in question operates at 33% and 67% of normal capacity in the first two years, respectively, and 100% of normal capacity thereafter. (In general, normal capacity is equivalent to 300 days per year operation at design capacity.)

TABLE 3.8.5. Assumed Ownership of Reference Fuel Cycle and Waste Management Facilities

Facility	Ownership
Nuclear power plant	Public utility
Spent fuel storage basin and related facilities, including spent fuel packaging facility	Large private company or federal government
Fuel reprocessing plant and related waste management facilities	Large private company
Mixed oxide fuel fabrication plant and related waste management facilities	Large private company
Centralized interim waste storage facilities	Federal government
Transportation equipment	Large private company
Federal repository (geologic isolation)	Federal government
Centralized plutonium oxide storage facility	Federal government

3.8.5.1 Levelized Cost Equations

The levelized cost equations for this report are based on the following relationship: the sum of present-worth revenue equals the sum of present worth expenditures, where expenditures include both capital and operating charges and the discount rate is the cost of capital. This relationship is expressed in the equation:

$$\sum^{n} pwf_n \text{ (revenue)}n = \sum^{n} pwf_n \text{ (expenditures)}n \qquad (1)$$

Where:

pwf_n = present worth factor in year n.

Satisfying this relationship means that the investment has been recovered and a specified return has been realized on the unrecovered investment over the indicated time period.

In this study, Equation 1 is the basis of two other equations, which take into account the weighted cost of equity and debt financing, state income and revenue taxes, property taxes and insurance, investment credits and federal surtaxes. These equations are used to determine a levelized unit capital charge (Equation 2) and a levelized unit operating charge (Equation 3) so that the relative contribution of the two types of charges to the total unit cost can be determined. These equations are:

$$LC_{Capital} = \frac{\sum^{n} pwf_n \left[C_n + WC_n - DEPN_n(T) - IC_n + TXINS_n\ (1-T)\right]}{\left[\sum^{n} pwf_n\ (UNITS)_n\right](1-T)} \qquad (2)$$

and

$$LC_{Operating} = \frac{\sum^{n} pwf_n\ (OE)_n}{\sum^{n} pwf_n\ (UNITS)_n} \qquad (3)$$

where

LC = Levelized unit cost

pwf_n = Present worth factor in year n using the cost of capital as the discount rate

C_n = Capital expenditure in year n

WC_n = Working capital in year n

OE_n = Operating expenses in year n

$DEPN_n$ = Depreciation in year n

$UNITS_n$ = Units produced in year n

IC_n = Investment credit in year n

$TXINS_n$ = Annual property tax and insurance (obtained as a percent of capital)

T = Weighted average federal and state income tax rates

= FITR + SITR (1-FITR)

where

FITR = Federal income tax rate

SITR = State income tax rate

3.8.5.2 Cost-of-Money Estimates

Cost-of-money estimates involve a number of assumptions, and a large uncertainty is inherent. This study employs an estimated, constant-dollar weighted cost of equity and debt to represent the cost of money. The estimated cost of equity capital and the estimated cost of debt capital are discussed separately, then combined as a single weighted average.

The cost of money can be considered to comprise three parts: 1) a riskless rate of return, 2) an inflation premium, and 3) a risk premium. The sum of these three components represents the "current-dollar," or inflated, cost of money. Both equity and debt costs include these components, although the magnitude of the components and totals will be different. Rates of return on equity and interest rates on borrowed capital (debt), based on published data, represent average current-dollar rates. To apply these data in a "constant-dollar" analysis, such as that developed here, a reduction for inflation effects must be considered. An increase in the risk premium for investments in radioactive waste management facilities is also appropriate.

Because of the high rates of inflation in recent years there has been a growing concern for the impact of inflation on reported profits and interest rates. The Financial Accounting Standards Board has been conducting field trials of accounting methods that use deflators to express financial data on a constant-dollar basis. The results indicate that net incomes for most companies are overstated - some by as much as 60%-70%.[6] It is apparent that current profits and interest rates are significantly overstated in real- or constant-dollar terms because of the effects of inflation. This was also true to a lesser extent even during the 1950s and early 1960s, a period of relatively low inflation. In spite of the current situation, there is no simple or uniformly accepted method of correcting interest rates to account for inflation effects.[7]

At the present time investments in the waste management area of the nuclear fuel cycle industry can be viewed as relatively high-risk ventures. Some of the factors contributing to the high risk include uncertainty regarding:

- the fuel cycle concepts that will prevail over the long term
- future government regulations controlling radioactive waste management functions
- public acceptance of specific waste management technologies.

If the industry develops as projected here, these uncertainties and the associated risk premium can be expected to decline as the industry matures.

An adjustment for inflation would reduce both the return on equity and the interest rates on debt for a constant-dollar analysis, while the addition of a larger risk premium would increase the rates. Rather than attempting to define and justify specific inflation and risk factors, this analysis assumes that the risk premium is equal to the current inflation premium, and current average cost-of-money rates are used. This assumption represents a relatively high risk premium and probably errs by overstating costs over the long period. The uncertainty in this assumption is recognized by considering a range of cost-of-money rates in evaluating overall unit cost uncertainties.

Cost of Equity Capital. Financial data on six large private companies with interests in the nuclear fuel cycle was used for estimating the required cost of equity and debt capital for facilities that would be privately owned. Because of the size of the capital investment and the expertise required, these companies are believed to be representative of those that might invest in nuclear waste management operations. Furthermore, since these companies are already engaged in nuclear activities, the overall risk and, hence, rate of return requirements perceived by investors, will probably not greatly change if the company makes investments in the nuclear waste management area. For these reasons, the rates of return on equity, the cost of debt and the capital structures of these companies are assumed to provide the appropriate basis for the private industry cost-of-money estimates. Return on equity data for these companies is shown in Table 3.8.6 for the period 1970-76. The average cost of equity capital for investments was 12% over this 7-yr period. A reference was adopted of 12% ± 5%, or a range of 7%-17%.

TABLE 3.8.6. Returns on Equity from 1970 to 1976 for Six Large Companies[8]

Company	Return on Equity, %						
	1970	1971	1972	1973	1974	1975	1976
Allied Chemical	6	7	8	10	15	11	10
Exxon	12	13	13	18	20	15	14
General Electric	13	17	17	17	16	17	18
Gulf	11	11	9	15	18	11	12
Union Carbide	9	9	11	14	21	14	14
Westinghouse	9	10	10	8	1	8	10
Average	10	11	11	14	15	13	13

A similar analysis is used to estimate the required cost of equity for an electric utility. The data in Table 3.8.7 show the average return on equity from 1970-76 for 6 public utilities in different parts of the U.S. The average return on equity assumed was about 10% ± 3%, or a range of 7%-13%.

TABLE 3.8.7. Returns on Equity from 1970 to 1976 for Six Electric Utilities[9]

Utility	Return on Equity, %						
	1970	1971	1972	1973	1974	1975	1976
Commonwealth Edison	11	10	11	11	9	9	10
Consolidated Edison	6	9	6	8	7	9	10
Duquesne Light	11	11	10	10	10	11	9
Florida Power	11	13	11	11	10	11	8
Northern States Power	10	10	10	10	9	11	11
San Diego Gas & Electric	10	10	9	8	10	6	10
Average	10	11	10	10	9	10	10

Cost of Borrowed Capital. Data on cost of debt capital for the same six private companies is shown in Table 3.8.8. The weighted-average cost of debt for these companies is 6.6%. However, the weighted cost of debt capital shown includes a significant number of debt issues at rates of 3%-4%. These lower-rate issues were undoubtedly made a number of years in the past. The average interest rate for the more recent issues has been about 8.5%. Since the companies analyzed will gradually incur new debt obligations to replace those that mature, the long-term cost of debt is estimated to be 8% ± 3%, or a range of 5%-11%.

TABLE 3.8.8. Cost of Debt for Six Large Companies[8]

Company	1976 Weighted-Average Cost of Debt, %	Recent Debt Issues	
		Year	Interest Rate, %
Allied Chemical	7.2	1975	9.0
Exxon	5.8	1968	6.5
General Electric	6.9	1974	8.5
Gulf	7.3	1970	8.5
Westinghouse	6.4	1970	8.625
Union Carbide	5.8	1975	8.5
Average	6.6		8.5

The weighted cost of debt capital for the six electric utilities is about 7%, as shown in Table 3.8.9. Most of the capital for these facilities is in more recent issues at the higher interest rates; hence, these rates are more indicative of the present cost of bonds. For this study, public utilities are assumed to have a cost of debt capital of 9% ± 2%, or a range of 7% - 11%.

TABLE 3.8.9. Cost of Debt for Six Electric Utilities with Nuclear Generating Capacity[9]

Utility	1976 Weighted-Average Cost of Debt, %	Recent Debt Issues	
		Year	Interest Rate, %
Commonwealth Edison	7.1	1976	9.0
Consolidated Edison	6.0	1975	9.5
Duquesne Light	7.1	1977	8.4
Florida Power	7.8	1975	9.4
Northern States Power	6.8	1975	9.5
San Diego Gas & Electric	7.6	1976	10.0
Average	7.1		9.3

Equity-Debt Ratios. The last item needed for the weighted cost-of-money estimates is the ratio of either debt or equity to total capital. Tables 3.8.10 and 3.8.11 show debt ratios for 1976 for the reference companies and utilities. For the private companies an equity-debt proportion of 75%-25% appears to be a reasonable average. For public utilities, a proportion of 50% debt and 50% equity appears reasonable.

TABLE 3.8.10. Debt Ratios for Six Large Companies[8]

	Debt Ratios						
	1970	1971	1972	1973	1974	1975	1976
Allied Chemical	0.34	0.36	0.34	0.32	0.30	0.37	--
Union Carbide	0.34	0.33	0.32	0.31	0.26	0.32	0.27
Exxon	0.18	0.19	0.18	0.16	0.16	0.16	--
Gulf	0.24	0.28	0.26	0.22	0.19	0.17	0.16
Westinghouse	0.22	0.29	0.27	0.25	0.25	0.30	0.23
General Electric	0.22	0.22	0.23	0.21	0.24	0.20	0.19
Average	0.27	0.28	0.27	0.25	0.23	0.25	0.21

TABLE 3.8.11. Debt Ratios for Six Electric Utilities[9]

Utility	Debt Ratios						
	1970	1971	1972	1973	1974	1975	1976
Florida Power	56	58	54	54	58	57	55
San Diego Gas & Electric	52	53	53	48	50	52	51
Duquesne Light	57	58	56	55	54	54	53
Northern States Power	48	50	50	51	53	52	50
Commonwealth Edison	55	53	49	50	47	51	52
Consolidated Edison	53	52	51	52	51	50	46
Average	54	54	52	52	52	53	51

Cost of Money for Government-Owned Facilities. The cost of money that may be used in establishing unit charges for federally-owned facilities will be determined administratively. It is assumed that services for any such facilities would be priced in a manner similar to that used for establishing uranium enrichment charges at the government-owned gaseous diffusion plants. These charges are currently (early 1978) based on full cost recovery and a 6.5% cost-of-money rate derived from the rate of interest on U.S. government securities. A 7% cost of money is assumed in this analysis, but in recognition of the uncertainty a range of 0%-10% is evaluated.

Weighted Costs of Money. Table 3.8.12 summarizes the estimated costs of equity and debt capital and the debt-to-capital ratios for private industry, utility, and government ownership of waste management facilities. The table also gives the after-tax, weighted costs of capital for these three types of ownership. The weighted costs were calculated using the equation:

$$\text{Cost of capital} = i_e(1-b) + i_b(b)(1 - T) \qquad (4)$$

where

i_e = Real rate of return on equity capital including risk premium

i_b = Real rate on debt capital including risk premium

b = Fraction of total capital derived from bonds (debt)

T = Weighted average tax rate for state and federal income taxes of 51.4%. (The assumed federal tax rate is 48%; the assumed state tax rate is 6.5%.)

The debt segment of this equation is adjusted to reflect the tax credit realized on the bond interest. (*Bond interest is not taxable; this, in effect, reduces the cost of capital.*)

TABLE 3.8.12. Weighted Cost of Money Estimates for Private, Utility, and Government Ownership of Waste Management Facilities

Ownership	Cost of Equity Capital, %	Cost of Debt Capital, %	Debt-to-Capital Ratio, %	Weighted Cost of Money, %
Private industry	12 ± 5	8 ± 3	25	10 ± 4
Electric utility	10 ± 3	9 ± 2	50	7 ± 2
Federal government	7^{+3}_{-7}	--	--	7^{+3}_{-7}

3.8.5.3 Financial Parameters in Levelized Cost Equations

Table 3.8.13 summarizes all of the parameters used in the levelized cost equations. State income and property tax rates were derived using average values from the set of all states levying these taxes.(10)

3.8.5.4 Effect of Economic Life on Fixed Charges

One of the variables affecting the calculated levelized unit cost is the economic life of the capital facility. The portion of the levelized unit cost necessary to recover capital and interest is called the fixed or capital charge; the ratio of total annual capital charges to capital investment is the fixed charge rate. In general fixed charge rates, and consequently

the levelized unit charge, are reduced as the economic life is increased. However, a point is reached where increasing economic life has little effect on reducing the fixed charge rate.

TABLE 3.8.13. Reference Financial Parameters used for Deriving Levelized Unit Costs

Financial Parameters	Ownership		
	Electric Utility	Large Private Company	Government
Method of depreciation for tax calculation	Sum of years digits	Sum of years digits	--
Federal income tax rate	0.48	0.48	--
State income tax rate	0.065	0.065	--
Investment credit rate	0.07	0.07	--
Property tax, insurance and contingencies	0.07	0.07	--
Equity rate of return	0.10 ± 0.03	0.12 ± 0.05	$0.07 \begin{smallmatrix} + 0.03 \\ - 0.07 \end{smallmatrix}$
Bond interest rate	0.075 ± 0.02	0.08 ± 0.03	--
Ratio of debt to total capital	0.50	0.25	--
Weighted cost of money	0.07 ± 0.02	0.10 ± 0.04	$0.07 \begin{smallmatrix} + 0.03 \\ - 0.07 \end{smallmatrix}$

The net effect of economic life on fixed charges for private and government ownership is illustrated in Figures 3.8.1 and 3.8.2, respectively. Both figures show the fixed charge rates as functions of plant life for three different costs of capital. The required annual fixed charge rates drop very sharply over the first ten years and then begin to level off. For either private or federal ownership, the fixed charge rates decline very slowly after 15 or 20 years. However, fixed charge rates for government ownership are somewhat more sensitive to economic life. The rates differ between private and federal ownership for the same cost of capital due to the effect of taxes. At higher costs of capital, a relatively constant charge rate is reached at a shorter plant life. For this study, most unit costs are based on a 15-year economic plant life. The text notes when plant lives other than 15 years are used, as in some of the storage facilities. The overall net effect of plant life on unit costs is further diluted by the contribution of operating charges to the total unit cost.

Many of the process facilities considered in this analysis could be expected to remain in service longer than 15 years. In fact, a service life of 30 years is assumed for most facilities. However, private industry does not generally consider a planning period for recovery of investments that is longer than 15 years. Basing the cost calculations on a 15-year life is consistent with private industry practices, and as discussed above, this shorter economic life does not significantly affect the unit cost. Furthermore, maintenance and equipment replacement requirements tend to increase with facility age and this would tend to counterbalance the fixed charge reduction. Although it is not anticipated, the entire facility could be replaced after 15 years with no increase in unit costs (in constant dollars) beyond those estimated here.

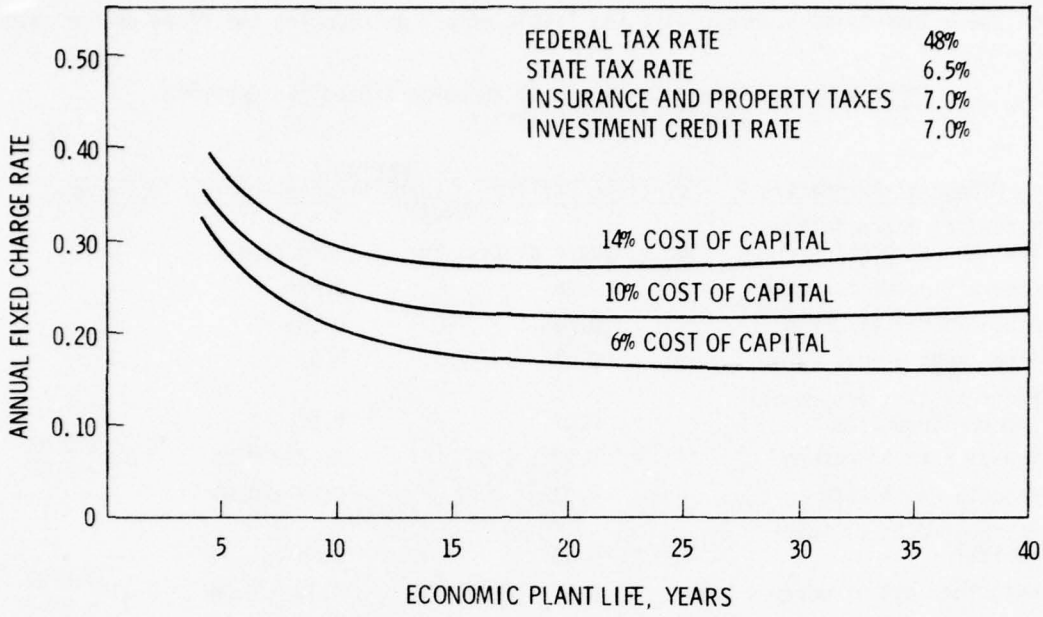

FIGURE 3.8.1. The Effect of Plant Life on Fixed Charges for Private Ownership

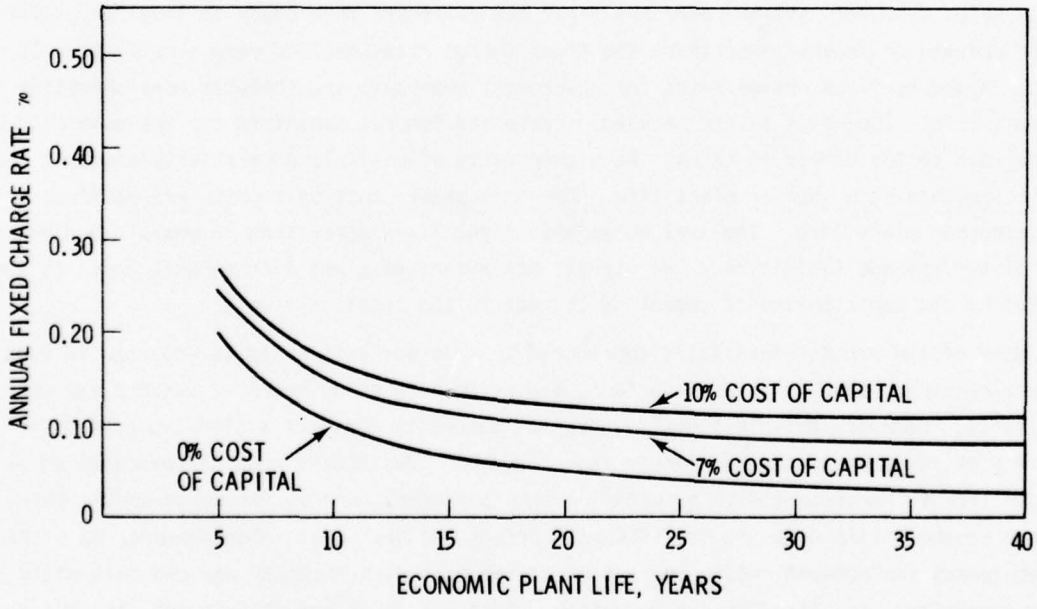

FIGURE 3.8.2. The Effect of Plant Life on Fixed Charges for Federal Government Ownership

3.8.6 Uncertainty Ranges for Cost Calculations

Uncertainties in the levelized unit cost estimate include contributions from three components: 1) capital costs, 2) operating costs, and 3) the cost of money; uncertainty ranges have been developed for each of these three components. The range for capital costs reflects uncertainties in the definition of the engineering scope required to provide a fully-functional plant based on the technology described, as well as uncertainties in the pricing and quantities for labor, materials, and equipment. A contingency covering these and similar factors has been included in the base capital cost estimate. With the contingency included, there is an approximately equal likelihood of an overrun or underrun of the indicated cost. The uncertainty for capital costs ranges from about ±20% to ±45%, depending on the facility and equipment, with a median uncertainty of about ±30%. For most facilities, uncertainty in the operating costs is estimated to range from +50% to -25%. In those few cases where a single cost component far exceeds all other operating costs (e.g., the cost of stainless steel canisters and baskets for packaged spent fuel water basin storage), more consideration was given to that cost component; in such cases the uncertainties may be smaller (e.g., ±25%). Because of the capital-intensive nature of the industry, the capital charge uncertainty generally overshadows the operating cost uncertainty for most of the facilities evaluated. Cost of money uncertainty ranges were identified in Table 3.8.12.

It was highly unlikely that the maximum uncertainty for all three components would ever coincide in either the plus or minus direction. Thus, it would be inappropriate to define the total unit cost uncertainty as the sum of the component uncertainties. Instead, the most probable range of uncertainty was approximated by first identifying the component with the largest absolute uncertainty; this was usually the capital cost. Then, 50% of the absolute uncertainty contributed by each of the remaining two components was added. A statistical analysis of several example unit cost calculations, assuming a normal random distribution of uncertainty around the three variables, indicates that the results approximate a 95+% confidence-level total uncertainty in the levelized unit cost. The following example shows how the uncertainty ranges were developed:

The levelized costs for the high-level waste calcination facility were calculated to be:

Levelized capital charge	=	$ 8.20/kg
Levelized operating charge	=	$ 2.90/kg
Total levelized cost	=	$11.10/kg

The uncertainty ranges for each of the three cost components expressed in terms of $/kg, were:

Capital cost uncertainty	=	±$ 2.50/kg
Operating cost uncertainty*	=	±$ 0.70/kg
Cost of money uncertainty	=	+$ 2.10/kg -$ 1.60/kg

Since the capital cost uncertainty was greatest, the total uncertainty was calculated as follows:

---

* In this example, canister costs are the controlling operating cost uncertainty.

Positive uncertainty = 2.50 + 0.50 (.70) + 0.50 (2.10) = +$3.90/kg

Negative uncertainty = -2.50 + 0.50 (.70) + 0.50 (-1.60) = -$3.65/kg

The positive and negative uncertainties were then expressed as percentages and rounded to the nearest 5%:

$$\text{Positive uncertainty} = \frac{3.90 \text{ \$/kg}}{11.10 \text{ \$/kg}} = +35\%$$

$$\text{Negative uncertainty} = \frac{-3.65 \text{ \$/kg}}{11.10 \text{ \$/kg}} = -35\%$$

In this example, as in most cases, the uncertainty range on capital cost dominates, although the cost of money also has a substantial effect on the overall uncertainty.

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3.9 BASIS FOR SAFEGUARDS AND PHYSICAL PROTECTION REQUIREMENTS

3.9 BASIS FOR SAFEGUARDS AND PHYSICAL PROTECTION REQUIREMENTS

Safeguards, including physical protection, are measures employed to prevent the theft or diversion of special nuclear material, to prevent the willful release of radioactive material and to prevent the sabotage of nuclear facilities. Facilities specifically designed for management of TRU wastes, including spent fuel and other highly radioactive nuclear wastes from the commercial nuclear power cycle, have yet to be licensed. However, implementation of the regulations already in place to protect the public from theft of nuclear material and from sabotage at licensed nuclear facilities serves also to protect the public during TRU and other radioactive waste management operations at these facilities. New facilities dedicated to waste management operations will also be subject to the regulations as well as any special conditions established in the licensing process.

Theft of TRU wastes containing the fissile material in strategic quantities is of greater concern than sabotage since the motive for theft could be to threaten the public with dispersal of the materials or with the assembly and explosion of a nuclear device. The consequences of sabotage, on the other hand, would be confined mainly to the environs of industrial sites dedicated to nuclear activity.

3.9.1 The Threat

The Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC) systematically collect and analyze information characterizing the perceived threat of nuclear theft and sabotage. Summaries of the threat are available in publications WASH-1535,[1] NUREG-0116,[2] NUREG-0095,[3] and the Code of Federal Regulations Title 10, Part 73.[4] A threat definition proposed by the NRC as one of the bases for general performance requirements for a physical protection system was accepted in developing the conclusions reached in this report. The threat is defined as:[5]

> (1) - A determined violent external assault, attack by stealth, or deceptive actions, by a small group with the following attributes, assistance and equipment: (i) well-trained (including military training and skills) and dedicated individuals, (ii) inside assistance which may include a knowledgeable individual who attempts to participate in both a passive role (e.g., provide information) and an active role (e.g., facilitate entrance and exit, disable alarms and communications, participate in violent attack), (iii) suitable weapons, up to and including hand-held automatic weapons, equipped with silencers and having effective long range accuracy, (iv) hand-carried equipment, including incapacitating agents and explosives for use as tools of entry or otherwise destroying the plant or transport integrity, and (v) the ability to operate as two or more teams, (2) An insider, including an employee (in any position), and (3) A conspiracy of insiders or employees in any position.

The motives of an adversary for a postulated nuclear waste incident are a matter of speculation. They might include revenge, blackmail, extortion, ransom, publicity, or mass destruction. They may be highly irrational. For further discussion of the threat, readers should see References 1 and 2 and the bibliographies given in those references.

3.9.2 Targets and Their Characteristics

The possible targets for theft or sabotage in the waste management systems of the light water reactor fuel cycle are assumed to be the plutonium and fission products in nuclear wastes in processes, in storage, and in transport. The likelihood of an attempt to steal wastes containing special nuclear material or highly radioactive material or to sabotage a facility will be affected significantly by two characteristics of the targets: the attractiveness and the accessibility of the wastes.

Two key elements of the attractiveness of wastes are its composition and form. Adversaries are attracted to the plutonium and fission products contained in the wastes. Plutonium is very dilute in most wastes and in some is accompanied by large quantities of radioactive fission products. However, some relatively pure plutonium compounds are stored as waste. These would be very rigorously safeguarded to prevent diversion of sufficient material for an explosive device. To isolate the plutonium from other wastes would require chemical processing that would be prohibitively expensive and time consuming for the adversary. In most cases, processing might involve the risk to adversaries of exposure to high levels of radiation from large amounts of accompanying fission products. If theft of plutonium for assembly of a weapon were the objective, separation of plutonium would be required for 1) spent fuel, containing up to 2% plutonium and 3% fission products, 2) high-level waste, containing nearly all the fission products of the process feed with more than 99% of the plutonium removed (in the case of uranium-only recycle, however, the high-level waste may contain all of the plutonium), and 3) waste with intermediate to extremely small amounts of fission products and small amounts of plutonium.

The form of waste is another element of its attractiveness; namely, is it readily or easily dispersible over large land areas or into waterways and is it in a form that is a health hazard to the public. Wastes that are highly radioactive but contain too little strategic material for eventual assembly of weapons may be in a form for possible dispersal into the environment and may be potential threats. Similarly, wastes containing plutonium in quantities that may be a threat to health if dispersed may also be attractive to an adversary. Although the actual health threat may be acceptably small in the case of small amounts of radioactive materials, they must be safeguarded because the psychological impact on the public may be significant were they to be dispersed.

An adversary's desire to sabotage or disperse TRU and other wastes is diminished because such acts would not be spectacular compared to an explosion and because the effects of radiation would usually not pose an immediate threat. The results of exposure would be evident among the public only after many years.

The second key feature affecting safeguard requirements for nuclear wastes is accessibility. Factors affecting accessibility include: 1) quantity available at a given location, 2) the degree of isolation of the location, 3) the complexity of the devices necessary for handling the material (e.g. whether they are operated manually or automatically and whether special knowledge or skills are required), and 4) the special health and safety measures required in handling the material.

The probability that hazards to the public will be caused by incidents resulting from theft of waste materials or sabotage of waste handling facilities is expected to be extremely low because of low attractiveness, low accessibility, and/or a highly effective safeguards system. These three elements can be related to the factors in the following equation which identifies the risk to society.

Risk to Society = Frequency x Success Rate x Consequences

This relationship indicates that the frequency of attempts, related in part to the attractiveness of the materials; a probable success rate, related in part to the availability of the material; and consequences, measured by effects on the public and the environment, all contribute to the risk to society. If one or more of these elements is small, the risk to society is also small. Realistic frequency and success probabilities are often difficult to develop. They depend upon the adversary's motivation, skill, and financial resources as well as the attractiveness and availability of material. A reliable consequence factor is generally more readily determined. However, safeguards measures for the waste are expected to reduce the probabilities of frequency and success to very small values and the subsequent risk to society to acceptable levels. Safeguards will also significantly reduce the consequence factor by including emergency contingency plans to minimize the effects of threats on the public, either real or perceived.

Because it is often covered by safeguards activities established for other materials or because it is not itself an available or attractive target, commercial nuclear power cycle wastes, with the exception of plutonium oxide discards, seem less of a threat for theft and sabotage than other materials in the LWR fuel cycle. The overall unattractiveness and limited availability of most forms of nuclear waste has led some investigators to dismiss the risk of sabotage and theft.[6]

3.9.3 Physical Protection and Safeguards Contingency Plans

Physical protection requirements for plants and materials are specified in the Code of Federal Regulations. The principal features of these requirements include protection forces (guards), physical and procedural access controls, detection aids, communication systems, and plans for emergencies.[4] Equipment items, systems, devices, or materials whose failure, destruction, or release could directly or indirectly endanger the public health and safety by exposure to radiation are considered "vital," and such equipment is subject to specific protection measures. The site-specific identification of "vital equipment and material" is a necessary part of the review of the physical protection plan submitted by an operating license applicant. Certain wastes would be classified as vital material and therefore would be located

in areas protected by two barriers which control access. Similarly, certain equipment needed for protecting wastes, such as cooling equipment for tanks of high-level liquid waste (HLLW), probably would be considered vital. Waste within the vital areas of fuel reprocessing plants and nuclear power plants and in the vital areas of facilities licensed to use plutonium would be subject to the same protection against sabotage and theft afforded vital equipment and material. Plutonium in transit is also covered by provisions in 10 CFR 73, and further guidelines for physical protection in shipping strategic nuclear materials are available in several regulatory guides.[9,10,11] Irradiated fuel elements in transit are exempt from these requirements, provided they give a dose rate in excess of 100 rem/hr at 0.9 m without shielding.[4] Recommendations for protecting wastes at power plants against industrial sabotage are specified in Regulatory Guide 1.17[12] and ANSI N18.17-1973.[13] Some waste that is not vital material or is not located in a vital area of a facility may not come specifically under the provisions specified in the current regulations. However, licensees of nuclear power reactors and those licensed to possess strategic amounts of plutonium, U-233 or U-235, must have and be capable of implementing plans for responding specifically to threats, thefts and attempts at industrial sabotage of licensed nuclear materials and facilities.[14,15]

Further guidance from the NRC for physical protection and safeguarding of nuclear materials is available in various regulatory guides.[16-21]

Other safety and environmental control features at nuclear reactors and plants licensed to have significant quantities of strategic nuclear materials also contribute significantly to safeguarding this material. Examples are:

- shielding of HLLW tanks
- tornado, earthquake, and flood protection of plutonium processing plants
- shielding (walls up to 2 m thick) for nuclear power plants and most nuclear reprocessing plant areas
- monitored cells or fuel cycle operations equipment built to detect and cope with release of radioactive materials
- plant fire and explosion protection
- shielding with 100-ton casks during transit, which deters sabotage and mitigates its potential impact
- tight packing of irradiated fuel elements in shipping casks, making removal difficult
- securing containers to shipping vehicles
- requiring security at transportation terminals
- remote locations for waste storage.

Licensees are required to meet NRC's license requirements for the above measures.

3.9.4 Specific Facility Protective Measures

Appropriate safeguards and physical protection measures are described for each of the waste treatment, transport, storage and disposal alternatives in the subsequent sections of this report. These measures are based upon consideration of the prospective threat, attractiveness, and accessibility of the waste and safeguards that are applicable for each type of waste process and facility.

REFERENCES FOR SECTION 3.9

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3.10 FUEL CYCLE PROJECTIONS

3.10 FUEL CYCLE PROJECTIONS

An assessment of the potential requirements for managing spent fuel, high-level and other TRU wastes requires a nuclear power growth projection as a basis. This section describes the nuclear power growth assumptions, the spent fuel quantities discharged, and the number and schedule of primary fuel cycle facilities required, for the once-through and the reprocessing fuel cycle cases described in Section 3.1. More detailed tables of the fuel cycle mass flows and inventory accumulations for these cases as well as the deferred decision and delayed repository cases are presented in the waste management system analysis in Section 10.

An assumption for the reference nuclear power growth projection is that nuclear generating capacity, all light water reactors (LWRs), will reach 400 GWe in the year 2000. No additional nuclear power plants starting up after that year are considered. This is not intended to imply that additional nuclear power capacity may not be commissioned after the year 2000. Rather, the intent is to develop a comprehensive assessment of the waste management commitment implied by the projected year 2000 system operating through its useful life. All nuclear power plants are assumed to operate for a 40-year lifetime, after which they are decommissioned. Thus, the installed generating capacity for the system considered here is reduced to zero in the year 2040. The generating capacity growth projection to the year 2000 for this reference case is slightly higher than current DOE maximum estimates, but it corresponds generally to those projections.[1,2]

The reference nuclear power growth projection, the the electric energy generated, and the annual and cumulative total spent fuel discharges are shown in Table 3.10.1 Each nuclear power plant is assumed to start up at 40% of capacity, increase to 70% in the fourth year, operate at 70% for 22 years and then decline linearly until the plant shuts down at 40% of capacity in its fortieth year. The resulting average system cpacity factor is also shown in Table 3.10.1. The system is composed of one-third BWRs and two-thirds PWRs and all plants are assumed to attain the startup and equilibrium core fuel exposures shown in Table 3.3.2.

Waste management system analyses based on this reference nuclear power growth projection are developed in Section 10 for all of the fuel cycle options described in Section 3.1. In addition, for comparison, basic once-through cycle and uranium-plutonium recycle cases are developed for a low-growth projection. This low-growth projection, shown in Table 3.10.2, is based on DOE's current low-range estimate for nuclear power growth.[2]

The reference once-through cycle (Case 1), has a substantial requirement for unpackaged spent fuel storage in conventional water basins to provide the required 6.5 year cooling prior to packaging (encapsulation) for disposal. An estimated 75% of this requirement could be met by the nuclear power plant basins. The storage requirements for Case 1 are shown in Table 3.10.3. These storage inventories exclude the first 6-month inventory that is required at power plant basins for operational purposes no matter what path the post-fission fuel cycle takes. The compatible storage requirements for the low-growth projection are shown in Table 3.10.4.

TABLE 3.10.1. Reference Nuclear Power Growth Projections

Year	Nuclear Capacity GWe[a]	Average Capacity Factor,%[b]	Electric Energy Generated 10^9 kWh[b]	Spent Fuel, MTHM	
				Annual Discharges	Cumulative Discharges
1977	36.1	62.2	178	700	1,780
1980	60.1	67.5	344	1,270	6,930
1985	127	65.9	682	2,480	16,300
1990	194	67.8	1,110	4,070	38,700
1995	293	67.7	1,670	6,210	60,200
2000	400	67.9	2,320	8,740	98,600
2005	399	68.6	2,400	9.090	144,500
2010	395	66.6	2,310	9,240	190,000
2015	364	65.2	2,100	8,600	235,200
2020	340	60.7	1,820	7,720	275,500
2025	274	55.4	1,370	7,150	313,600
2030	206	49.4	923	5,310	343,000
2035	108	43.4	450	3,960	356,900
2040	0	30.4	27	1,530	379,400

a. At year end.
b. During the year.

TABLE 3.10.2. Low Nuclear Power Growth Projections

Year	Nuclear Capacity GWe[a]	Average Capacity Factor,%[b]	Electric Energy Generated 10^9 kWh[b]	Spent Fuel, MTHM	
				Annual Discharges	Cumulative Discharges
1977	36.1	62.2	178	700	1,780
1980	60.1	67.5	344	1,270	6,930
1985	100	66.9	560	2,060	15,300
1990	157	67.1	875	3,260	28,800
1995	200	68.7	1,180	4,330	48,700
2000	255	68.0	1,490	5,700	74,200
2005	254	67.7	1,510	5,660	103,200
2010	250	65.3	1,440	5,910	132,100
2015	219	64.4	1,260	5,300	160,700
2020	195	59.4	1,020	4,570	185,000
2025	155	54.2	760	3,950	206,600
2030	98	48.9	450	3,210	224,200
2035	55	43.8	230	1,830	234,900
2040	0	30.0	15	850	241,800

a. At year end.
b. During the year.

TABLE 3.10.3. Spent Fuel Storage Requirements for the Reference Once-Through Cycle

Year	Unpackaged Spent Fuel Storage, MTHM		
	LWR Basins[a]	Independent Basins	Total
1980	6,300	0	6,300
1985	12,900	1,500	13,400
1990	15,700	5,200	20,900
1995	22,000	7,300	29,300
2000	32,400	10,800	43,200
2005	40,700	13,600	54,300
2010	41,000	13,700	54,700
2015	40,900	13,600	54,500
2020	37,200	12,400	49,600
2025	34,700	11,600	46,300
2030	28,200	9,400	37,600
2035	21,700	7,200	28,900
2040	14,000	4,700	18,700
2045	2,000	700	2,700
2047	0	0	0

a. Not included in this inventory is the amount of spent fuel representing the first 6-month storage.

TABLE 3.10.4. Spent Fuel Storage Requirements for the Low Growth Once-Through Cycle

Year	Unpackaged Spent Fuel Storage, MTHM		
	LWR Basins[a]	Independent Basins	Total
1980	6,300	0	6,300
1985	12,300	1,300	13,600
1990	12,900	4,300	17,200
1995	16,900	5,600	22,500
2000	21,800	7,300	29,100
2005	25,900	8,600	34,500
2010	25,800	8,600	34,400
2015	26,000	8,700	34,700
2020	22,600	7,500	30,100
2025	19,900	6,600	26,500
2030	16,500	5,500	22,000
2035	11,100	3,700	14,800
2045	1,100	400	1,500
2047	0	0	0

a. Excludes the first 6-month storage in the power plant basins.

Spent fuel storage requirements are substantially reduced for the reference spent-fuel reprocessing cycles (Cases 2 and 3). Reprocessing is assumed to start in 1981 at the 1500 MTHM/yr, Allied General Nuclear Services (AGNS) plant in Barnwell, SC. Subsequent fuel reprocessing plants (FRPs) are assumed to be 2000 MTHM/yr FRPs as described in Section 3.2. These are added to the system in 1985, 1993, 1998 and 2003. Two additional plants are added in 2012 and 2020 as replacements for earlier plants that were shut down after an assumed 30 year useful life. A reduced-capacity 2-year startup period was assumed for each plant during the first and second year of operation when the plants operate at one-third and two-thirds capacity, respectively. The spent fuel storage requirements, reprocessing rates and mixed oxide (MOX) fuel fabrication rates (Case 3 only) for the reference reprocessing cases are shown in Table 3.10.5. The comparable data for the low-growth projection are shown in Table 3.10.6. With these early reprocessing dates, independent storage basins do not appear to be needed. However, for deferred reprocessing (Case 4b) substantial independent storage capacity is required. This is described in Section 10.

TABLE 3.10.5. Spent Fuel Storage Requirements, Reprocessing and MOX Fuel Fabrication Rates for the Reference Reprocessing Cases

Year	Spent Fuel Storage, MTHM[a]	Reprocessing Rate, MTHM/yr	MOX-Fuel Fabrication Rate MTHM/yr
1980	6,300	0	0
1985	8,400	2,170	400
1990	8,100	3,500	860
1995	12,100	5,500	1,460
2000	17,700	7,500	1,970
2005	21,900	9,500	2,580
2010	19,900	9,500	2,850
2015	21,400	8,000	2,830
2020	21,600	8,760	2,550
2025	17,000	8,000	2,240
2030	13,400	6,000	1,480
2035	12,900	4,000	470
2040	7,600	4,000	0
2044	0	355	0

a. Excludes the first 6-month storage in the power plant basins.

TABLE 3.10.6. Spent Fuel Storage Requirements, Reprocessing and MOX Fuel Fabrication Rates for the Low-Growth Reprocessing Cases

Year	Spent Fuel Storage, MTHM[a]	Reprocessing Rate, MTHM/yr	MOX-Fuel Fabrication Rate MTHM/yr
1980	6,300	0	0
1985	8,300	1,500	330
1990	7,700	3,500	840
1995	9,500	3,500	960
2000	12,900	5,500	1,510
2005	14,400	5,500	1,560
2010	13,700	6,830	1,990
2015	12,500	6,000	1,800
2020	15,200	4,000	1,280
2025	17,100	4,000	1,170
2030	19,100	3,330	690
2035	10,500	4,000	250
2040	1,920	2,000	0
2042	0	655	0

a. Excludes the first 6-month storage in the power plant basins.

REFERENCES FOR SECTION 3.10

1. USDOE's Energy Information Administration, Annual Report to Congress, Volume II, "Projections of Energy Supply and Demand and Their Impacts," DOE report DOE/EIA-0036/2, April, 1978.

2. Letter from R. Gene Clark, Chief, Nuclear Energy Analysis Division of DOE to Mel W. Shupe, DOE, Richland Operations Office, July 12, 1978.

APPENDIX 3A

POSTULATED ACCIDENT LIST AND UMBRELLA SOURCE TERM INDEX

APPENDIX 3A

POSTULATED ACCIDENT LIST AND UMBRELLA SOURCE TERM INDEX

This Appendix contains the title, accident number, and reference umbrella source term identification for all postulated accidents for all waste management functions described in this report. The methodology used to identify potential accidents, determine their severity classification, and select an appropriate umbrella source term* for environmental consequence analysis is presented in Section 3.7 - Basis for Accident Analysis. The environmental consequences of the postulated accidents are presented in DOE/ET-0029.

The Appendix is divided into two lists. The first presents the number and title of all postulated accidents. The number in the second column in this list corresponds to the umbrella source term related to each accident. The worst case source term (see Section 3.7 for criteria) for a group of related accidents is designated as an umbrella source term.

The second list is a cross index of all accidents under each umbrella source term accident. These are grouped by accident severity classification.

Accidents with potential to significantly increase radiation fields in an occupied or potentially occupied zone are designated by a footnote. These accidents may result in occupational exposures.

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*For a definition of the umbrella source term, see page 3.7.1

APPENDIX 3A - PART 1

POSTULATED ACCIDENT LIST*

	Postulated Accident	Umbrella Source Term
4.1	**High-Level Waste Solidification**	
4.1.1	**High-Level Waste Vitrification**	
	Minor	
4.1.1	Hydrogen explosion in feed tank	4.1.3
4.1.2	HLLW feed system leakage	4.1.3
4.1.3	Calcine spill due to backup in equipment or other process irregularity	4.1.3
4.1.4	Sintered metal filter failure	4.1.3
4.1.5	Canister and retort failure during melting operation	4.1.3
	Moderate	
4.1.6[(a)]	Feed solution backup in air line or contamination spread to occupied zones	4.1.6
4.1.7	Off-gas iodine or ruthenium adsorber failure	4.1.8
4.1.8	Process off-gas filter failure	4.1.8
4.1.9[(a)]	Off-gas blower failure	4.1.6
4.1.10	Failure of cell exhaust filters	4.1.8
4.1.2	**High Level Waste Calcination**	
	Minor	
4.1.11	Hydrogen explosion in feed tank	4.1.3
4.1.12	HLLW feed system leakage	4.1.3
4.1.13	Calcine spill from calcine handling equipment due to process irregularity	4.1.3
4.1.14	Overheating of calciner equipment due to plugging	4.1.3
4.1.15	Calcine overheating in canister	4.1.3
	Moderate	
4.1.16[(a)]	Feed solution backup in air line or contamination spread to occupied zone	4.1.6
4.1.17[(a)]	Calciner pressurization due to malfunction of fuel ignition system	4.1.6

a. Accident with potential for increased worker exposure

* As an example of the use of this list, the estimated source term for Accident 4.1.3 is higher than that for any of these accidents: 4.1.1, 4.1.2, 4.1.4, 4.1.5, and 4.1.11 thru 4.1.15.

Postulated Accident		Umbrella Source Term
	Moderate	
4.1.18	Failure of off-gas filter or scrubber	4.1.8
4.1.19[a]	Loss of off-gas system flow	4.1.6
4.1.20	Ruthenium sorber failure	4.1.8
4.1.21[a]	Fire in cell	4.1.6, 4.1.8
4.1.22	Failure of cell exhaust filters	4.1.8
4.2 Packaging of Fuel Residue		
4.2.1 Fuel Residue Packaging without Compaction		
	Minor	
4.2.1	Zr fines fire	4.2.2
4.2.2 Mechanical Compaction of Hulls		
	Minor	
4.2.2	Zr fines fire	4.2.2
4.2.3 Hulls Melting Process		
	Minor	
4.2.3	Molten Zr explosion/fire	4.2.2
4.3 Failed Equipment and Noncombustible Waste Treatment		
4.3.1 Failed Equipment and Noncombustible Waste Treatment at a Fuel Reprocessing Plant		
4.3.2 Failed Equipment and Noncombustible Waste Treatment at a Mixed Oxide Fuel Fabrication Plant		
	Minor	
4.3.1	Failed equipment tip over during disassembly	No release
4.3.2	Failed equipment drop from crane	No release
4.4 Combustible and Compactable Waste Treatment		
4.4.1 Incineration Process for Intermediate-Level Waste (ILW) at the Fuel Reprocessing Plant		
4.4.2 Incineration Process for Low-Level Waste (LLW) at the Fuel Reprocessing Plant		
4.4.6 Incineration Process at a Mixed Oxide Fuel Fabrication Plant		

a. Accident with potential for increased worker exposure

Postulated Accident		Umbrella Source Term
	Minor	
4.4.1	Loss of cooling water to incinerator off-gas treatment system	No release
4.4.2	Minor fire in feed preparation system	4.1.3
	Moderate	
4.4.3[a]	Major fire in feed preparation line	4.4.3
4.4.4[a]	Explosion in feed preparation system	4.4.5
4.4.5	Incinerator explosion	4.4.5

4.4.3 Intermediate - Level Waste (ILW) Packaging Without Treatment at the Fuel Reprocessing Plant

4.4.7 General Trash Packaging Without Treatment at a Mixed Oxide Fuel Fabrication Plant

	Minor	
4.4.6	Ruptured waste bags spilled to floor	No release
4.4.7	Fire in barrel of bagged trash	4.4.8
4.4.8	Spent HEPA filters spilled to floor	4.4.8

4.4.4 Low Level Waste (LLW) Packaging Without Treatment at the Fuel Reprocessing Plant

	Minor	
4.4.6	Ruptured waste bags spilled to floor	No release
4.4.7[a]	Fire in barrel of bagged trash	4.4.8

4.5 Degraded Solvent Treatment

4.5.1 Degraded Solvent Incineration

	Minor	
4.5.1	Loss of electrical power	No release
	Moderate	
4.5.2	Incinerator flame-out explosion	4.4.5
4.5.3	Solvent fire	4.4.5
4.5.4	Explosion in incinerator	4.4.5

4.7 Immobilization of Wet and Solid Wastes

4.7.1 Bitumen Immobilization at an FRP

4.7.2 Cement Immobilization at an FRP

4.7.4 Bitumen Immobilization at a Mixed Oxide Fuel Fabrication Plant

4.7.5 Cement Immobilization at a MOX-FFP

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a. Accident with potential for increased worker exposure

Postulated Accident		Umbrella Source Term
	Minor	
4.7.1	Spillage during drum filling	No release
4.7.2	Drum filling control valve failure	No release
4.7.3	Drum filled level detector instrument failure	No release
4.7.4	Container drop or rupture	No release
4.7.5	Leakage in waste in transfer line	4.7.6
4.7.6	Bitumen fire	4.7.6
	Moderate	
4.7.7	Cell HEPA failure with any minor accident	4.7.8
4.7.8	Cell HEPA failure with bitumen fire	4.7.8

4.9 Fuel Reprocessing Plant Dissolver Off-Gas Treatment

4.9.1 Dissolver Off-Gas Iodine Recovery

	Minor	
4.9.1	Plugged iodine or ruthenium bed	No release
4.9.2	Torn HEPA filter[(a)]	
	Moderate	
4.9.3	Process shutdown while dissolver is operating	4.9.13
4.9.4	Iodine canister adsorbent spill during replacement	No release
4.9.5	Ruthenium canister adsorbent spill during replacement	No release

4.9.2 Dissolver Off-Gas Carbon-14 Recovery

	Minor	
4.9.6	Plugged recovery bed	No release
4.9.7	Plugged lime scrubber	No release
	Moderate	
4.9.8	Process shutdown with ^{14}C venting	4.9.14

4.9.3 Dissolver Off-Gas Krypton Recovery

	Minor	
4.9.9	Plugged lines[(a)]	
4.9.10	Freezing[(a)]	

a. All minor leakage is included in the operational release

Postulated Accident		Umbrella Source Term
	Moderate	
4.9.11	Process shutdown with Krypton-85 venting	4.9.14
4.9.12[b]	Oxygen recombiner explosion	5.6.4
4.9.4	Combined Dissolver Off-Gas System	
	Moderate	
4.9.13	Process shutdown with volatile venting - one day	4.9.13
4.9.14	Process shutdown with carbon-14 and krypton-85 venting--30 days	4.9.14
4.10	Process Off-Gas Treatment	
4.10.1	Vessel Off-Gas System at the Fuel Reprocessing Plant	
	Minor	
4.10.1	Plugged iodine bed	No release
4.10.2	Torn packaged glass fiber filter[a]	
	Moderate	
4.10.3	VOG Process Shutdown	4.9.13
4.10.2	Process Off-Gas Treatment at an ISFSB	
	Minor	
4.10.4	Plugged iodine bed	No release
4.10.5	Torn HEPA filter[a]	
	Moderate	
4.10.6	Process shutdown	4.9.13
4.10.7	Iodine canister adsorbent spill during replacement	No release
4.11	Fuel Reprocessing Plant Atmospheric Protection System	
4.11.1	Group III Filter Module/HEPA Filter Atmospheric Protection System	
4.11.2	Sand Filter/HEPA Filter Atmospheric Protection System	
4.11.3	Deep-bed Glass Fiber Filter/HEPA Filter Atmospheric Protection System	
	Minor	
4.11.1	Loss of normal electrical power	No release
4.11.2	Loss of exhaust fan	No release

a. All minor leakage is included in the operational release
b. Accident with potential for increased worker exposure

	Postulated Accident	Umbrella Source Term
	Moderate	
4.11.3	Rupture of final HEPA filters-extensive vibration and defective equipment	4.1.8
4.11.4	Rupture of HEPA filters - sudden increase in flow resistance	4.1.8
5.1	**High-Level Liquid Waste Storage**	
5.1.1	**Tank Storage of High-Level Liquid Waste**	
	Minor	
5.1.1	Loss of normal electrical power	No release
5.1.2	Leak from diversion equipment	No release
5.1.3	Temporary loss of sparge/purge air	No release
5.1.4	Loss of normal secondary coolant supply	No release
5.1.5	Temporary loss of exhaust blower	No release
	Moderate	
5.1.6	Filter fire	4.1.8
5.1.7	Breach of waste tank (primary container) with sound secondary liner	No release
5.1.8	Off-gas HEPA filter fails	4.1.8
5.2	**Storage of Fuel Residue**	
5.2.1	**Vault Storage of Fuel Residue**	
5.2.2	**Subsurface Storage of Fuel Residue**	
	Moderate	
5.2.1[(a)]	Waste hulls canister breached by drop	5.2.1
5.3	**Non-high Level Solid Waste Storage**	
5.3.1	**Outdoor Storage of Transuranic Low-Level Waste**	
	Minor	
5.3.1	Mechanical breach of drum	5.3.1
5.3.2	Dislodge of surface contamination	5.3.1
5.3.3	Overpressurization of sealed container	5.3.1
5.3.4	Container rusts through	5.3.1
	Moderate	
5.3.5[(a)]	Fire in storage stack	5.3.6
5.3.6[(a)]	Tornado strikes	5.3.6

a. Accident with potential for increased worker exposure

Postulated Accident		Umbrella Source Term
5.3.2	Indoor Unshielded Storage of Low Level Waste	
	Minor	
	See Accidents 5.3.1 - 5.3.4	
5.3.7[a]	Fire in storage stack	5.3.1
5.3.3	Outdoor Storage of Transuranic Intermediate - Level Waste	
	Minor	
	See Accidents 5.3.1 - 5.3.4	
	Moderate	
5.3.8[a]	Crane drops drum	5.3.8
5.3.4	Indoor Shielded Storage of Transuranic Intermediate - Level Waste	
	Minor	
	See Accidents 5.3.1 - 5.3.4	
	See Accident 5.3.7	
	Moderate	
	See Accident 5.3.8	
5.4	Solidified High-Level Waste Storage	
5.4.1	Water Basin Storage for Solidified High-Level Waste	
	Minor	
5.4.1	Loss of normal electrical power	No release
5.4.2	Loss of normal cooling water supply	No release
5.4.3	Reduction of normal cooling water supply	No release
5.4.4	Ventilation system failure	No release
5.4.5	Leak in water treatment systems	No release
5.4.6	Failure of secondary cooling loop	No release
	Moderate	
5.4.7[a]	Canister failure in storage basin	No release
5.4.8	Failure of basin liner	No release
5.4.9	Dropping shipping cask into cask well	No release
5.4.10[a]	Dropping canister into storage basin	No release
5.4.11	Failure of primary cooling loop	No release
	Severe	
5.4.12[a]	Concrete cover block falls into storage pool	No release
5.4.13[a]	Contamination of secondary cooling water	5.7.12
5.4.14[a]	Fire in storage basin building	No release
5.4.15	Design basis tornado	5.7.12

a. Accident with potential for increased worker exposure

	Postulated Accident	Umbrella Source Term
5.4.2	Sealed Storage Cask for Storage of Solidified High-Level Waste	
	Minor	
5.4.16	Receipt of externally contaminated shipping cask	No release
5.4.17	Receipt of contaminated or failed canisters	No release
5.4.18	Impaired waste cooling	No release
	Moderate	
5.4.19	Canister failure in receiving cell	5.4.19
5.4.20	Canister failure in weld and test cell	5.4.19
5.5	Interim Storage of Plutonium Oxide	
5.5.1	Interim Plutonium Oxide Storage Facility	
	Minor	
5.5.1	Loss of normal electrical power	No release
5.5.2	Temporary loss of ventilation blower	No release
	Moderate	
5.5.3	Decontamination trash fire	5.5.4
5.5.4	Storage container leakage	5.5.4
	Severe	
5.5.5	Storage container breach	5.5.5
5.5.6[(a)]	Plutonium product criticality	5.5.6
5.6	Krypton Storage	
5.6.1	Krypton Gas Cylinder Storage	
	Minor	
5.6.1	Loss of storage cell coolant	No release
	Moderate	
5.6.2	Kr cylinder ruptured in storage cell or hot cell	5.6.4
5.6.3	Kr cylinder corrodes	5.6.4
5.6.4[(a)]	Kr cylinder ruptured in operating area or storage corridor	5.6.4
5.7	Spent Fuel Storage	
5.7.1	Water Basin Storage of Unpackaged Spent Fuel	
5.7.2	Modified Water Basin Storage of Unpackaged Spent Fuel	

a. Accident with potential for increased worker exposure

	Postulated Accident	Umbrella Source Term
	Minor	
5.7.1	Loss of normal electrical power	No release
5.7.2	Loss of normal cooling water supply	No release
5.7.3	Reduction of normal cooling water supply	No release
5.7.4	Ventilation system failure	No release
5.7.5	Leak in water treatment system	No release
5.7.6[(a)]	Minor fuel handling accident	5.7.6
	Moderate	
5.7.7[(a)]	Fuel handling accident	4.1.6, 5.7.18
5.7.8[(a)]	Fuel handling accident without confinement protection	4.1.6, 5.7.18
5.7.9[(a)]	Rail cask venting	4.1.6, 5.7.18
5.7.10	Shipping cask dropped into cask unloading pool	No release
5.7.11[(a)]	Dropped fuel transfer basket	4.1.6, 5.7.18
	Severe	
5.7.12	Design basis tornado	5.7.12
5.7.13[(a)]	Criticality	5.7.13
5.7.14	Loss of cooling water supply	No release

5.7.3 Spent Fuel Packaging

5.7.4 Independent Packaged Spent Fuel Receiving Facility

	Minor	
5.7.15	Loss of normal electrical power	No release
5.7.16	Loss of normal cooling air supply	No release
5.7.17	Equipment failure - fuel assembly suspended in air	No release
	Moderate	
5.7.18[(a)]	Fuel handling accident	5.7.18
5.7.19	Shipping cask dropped into cask well	No release
	Severe	
5.7.20	Dropped fuel assembly - rupture 20% of the rods	5.7.20

a. Accident with potential for increased worker exposure

Postulated Accident		Umbrella Source Term
5.7.5	Water Basin Storage of Packaged Spent Fuel	
	Minor	
5.7.21	Loss of normal electrical power	No release
5.7.22	Loss of normal cooling water supply	No release
5.7.23	Reduction of normal cooling water supply	No release
5.7.24	Ventilation system failure	No release
5.7.25	Leak in water treatment system	No release
5.7.26	Dropped fuel rack in transfer tunnel	No release
	Severe	
5.7.27	Design basis tornado	5.7.12
5.7.28[(a)]	Criticality	5.7.13
5.7.29	Loss of cooling water supply	No release
5.7.6	Air Cooled Vault Storage of Packaged Spent Fuel	
	Minor	
5.7.30	Loss of cooling air flow	No release
5.7.31	Flooding of storage vault	No release
5.7.32	Dropped fuel rack in transfer tunnel or vault	No release
	Moderate	
5.7.33[(a)]	Packaged element fails in storage	5.7.39
5.7.7	Dry Caisson Storage of Packaged Spent Fuel	
	Minor	
5.7.34	Flooding of one caisson	No release
5.7.35	Dropped fuel rack in transfer tunnel or caisson	No release
	Moderate	
5.7.36[(a)]	Packaged element fails in storage	5.7.39
5.7.8	Surface Cask Storage of Packaged Spent Fuel	
	Minor	
5.7.37	Loss of cooling air flow	No release
5.7.38	Filled fuel rack dropped in transfer tunnel or storage cask	No release
	Moderate	
5.7.39[(a)]	Packaged element fails in storage	5.7.39

a. Accident with potential for increased worker exposure

Postulated Accident		Umbrella Source Term
6.0 Transportation of Nuclear Material		
6.2 Transport of Spent Fuel		
6.2.1 Rail Transport of Spent Fuel		
Minor		
6.2.1	Train derailment involves spent fuel cask	No release
6.2.2	Train derailment and 1/2 hour (or less) fire involves spent fuel cask	No release
6.2.3	Undetected leakage of coolant from cask cavity (and/or surface contamination washoff)	6.2.3
Moderate		
6.2.4	Collision or derailment results in loss of neutron shielding from spent fuel rail cask	6.3.4
6.2.5	Fire accompanying accident causes rail cask cavity to overpressurize; relief valve operates	6.2.6
6.2.6	Collision or derailment causes damage to rail cask mechanical cooling system	6.2.6
Severe		
6.2.7	Collision or derailment causes spent fuel rail cask to be subjected to severe impact and fire	6.3.5
6.2.8	Cavity coolant lost from spent fuel rail cask; no emergency action taken	6.2.8
6.2.2 Truck Transport of Spent Fuel		
Minor		
6.2.9	Truck collision or overturn accident involves spent fuel cask	No release
6.2.10	Truck collision or overturn accident and 1/2 hour (or less) fire involves spent fuel cask	No release
6.2.11	Undetected leakage of coolant from cask cavity (and/or surface contamination washoff)	6.2.3
Moderate		
6.2.12	Truck collision or overturn results in loss of neutron shielding from spent fuel truck cask	6.3.4
6.2.13	Fire accompanying accident causes truck cask cavity to overpressurize; relief valve operates	6.2.13

	Postulated Accident	Umbrella Source Term
	Severe	
6.2.14	Collision or overturn causes spent fuel truck cask to be subjected to severe impact and fire	6.3.5

6.3 Transportation of High-Level Waste

6.3.1 Rail Transport of High-Level Waste

	Minor	
6.3.1	Train derailment involves high-level waste cask	No release
6.3.2	Train derailment and 1/2 hour (or less) fire involves high-level waste cask	No release
6.3.3	Unusual transport condition erodes cask surface	6.2.3
	Moderate	
6.3.4	Collision or derailment results in loss of neutron shielding from solid high-level waste cask	6.3.4
	Severe	
6.3.5	Collision or derailment subjects high-level waste cask to severe impact and fire	6.3.5

6.4 Transport of Fuel Residues

6.4.1 Rail Transport of Fuel Bundle Residues

	Minor	
6.4.1	Derailment involves fuel residues cask	No release
6.4.2	Derailment and 1/2 hour (or less) fire involves fuel residues cask	No release
6.4.3	Unusual transport conditions erodes cask surface	6.2.3
	Severe	
6.4.4	Collision or derailment subjects fuel residues cask to severe impact and fire	5.2.1

6.5 Transportation of Plutonium

6.5.1 Truck Transport of Plutonium

	Potential accidents are similar to 6.2.2 accidents	No release

6.6 Transportation of Non-High-Level TRU Wastes

6.6.1 Truck Transport of Non-High-Level TRU Wastes

Postulated Accident		Umbrella Source Term
Minor		
6.6.1	Truck collision or overturn accident involves non-high-level TRU waste container	No release
6.6.2	Truck collision or overturn accident and 1/2 hour (or less) fire involves non-high-level TRU waste container	No release
6.6.3	Non-high-level TRU waste shipment made in improperly closed packages	6.6.3
Severe		
6.6.4	Non-high-level TRU waste container is subjected to severe impact and fire	6.6.4

7.0 Final Isolation and Disposal of Long Lived Wastes

7.5 Geologic Repositories for the Once Through Fuel Cycle

7.6 Geologic Repositories for the Reprocessing Fuel Cycle

Minor		
7.1[a]	LLW drum rupture due to handling error	4.4.8
7.2[a]	Minor canister failure	7.2
7.3	Receipt of externally contaminated canister	No release
7.4	Dropped shipping cask	No release
Moderate		
7.5[a]	Waste container drop during handling	7.6
7.6	Waste package dropped down mine shaft	7.6
7.7	Tornado strikes mined materials storage area	7.7
7.8[a]	LLW drum rupture due to mechanical damage and fire	7.6
7.9[a]	LLW drum rupture due to internal explosion	7.6
Non Design Basis		
7.10	Nuclear warfare	No release
7.11	Repository breach by meteorite	7.11
7.12	Repository breach by drilling	7.12
7.13	Repository breach by solution mining	7.13
7.14	Volcanism	7.15
7.15	Repository breach by faulting with ground water transport	7.15
7.16	Erosion	7.15
7.17	Criticality	No release

a. Accident with potential for increased worker exposure

APPENDIX 3A - Part 2

UMBRELLA SOURCE TERM INDEX

MINOR ACCIDENTS

4.1.3 Calcine spill due to backup in equipment or other process irregularity

- 4.1.1 Hydrogen explosion in feed tank
- 4.1.2 HLLW feed system leakage
- 4.1.4 *Sintered metal filter failure*
- 4.1.5 Canister and retort failure during melting operation
- 4.1.11 Hydrogen explosion in feed tank
- 4.1.12 HLLW feed system leakage
- 4.1.13 Calcine spill from calcine handling equipment due to process irregularity
- 4.1.14 Overheating of calciner equipment due to plugging
- 4.1.15 Calcine overheating in canister
- 4.4.2 Minor fire in feed preparation system

4.2.2 Zr fines fire

- 4.2.1 Zr fines fire
- 4.2.3 Molten Zr explosion/fire

4.4.8 Spent HEPA filters spilled to floor (FRP-ILW)

- 4.4.7 Fire in barrel of bagged trash
- 4.4.7[(a)] Fire in barrel of bagged trash
- 7.1[(a)] LLW drum rupture due to handling error

4.7.6 Bitumen fire

- 4.7.5 Leakage in waste in transfer line

5.3.1 Mechanical breach of drum

- 5.3.2 Dislodge of surface contamination
- 5.3.3 Overpressurization of sealed container
- 5.3.4 Container rusts through
- 5.3.7[(a)] Fire in storage stack

5.7.6[(a)] Minor fuel handling accident

6.2.3 *Undetected leakage of coolant from cask cavity (and/or surface contamination washoff)*

- 6.2.11 Undetected leakage of coolant from cask cavity (and/or surface contamination washoff)
- 6.3.3 Unusual transport condition erodes cask surface
- 6.4.3 Unusual transport condition erodes cask surface

---

a. Accident with potential for increased worker exposure

MINOR ACCIDENTS

6.6.3 Non-high-level TRU waste shipment made in improperly closed packages

7.2[a] Minor canister failure

MODERATE ACCIDENTS

4.1.6[a] Feed solution backup in air line or contamination spread to occupied zones

4.1.9[a] Off-gas blower failure

4.1.16[a] Feed solution backup in air line or cantamination spread to occupied zone

4.1.17[a] Calciner pressurization due to malfunction of fuel ignition system

4.1.19[a] Loss of off-gas system flow

4.1.21[a] Fire in cell

5.4.7[a] Canister failure in storage basin

5.4.10[a] Dropping canister into storage basin

5.4.12[a] Concrete cover block falls into storage pool

5.4.14[a] Fire in storage basin building

5.7.7[a] Fuel handling accident

5.7.8[a] Fuel handling accident without confinement protection

5.7.9[a] Rail cask venting

5.7.11[a] Dropped fuel transfer basket

4.1.8 Process off-gas filter or scrubber failure

4.1.7 Off-gas iodine or ruthenium adsorber failure

4.1.10 Failure of cell exhaust filters

4.1.18 Failure of off-gas filter or scrubber

4.1.20 Ruthenium sorber failure

4.1.21[a] Fire in cell

4.1.22 Failure of cell exhaust filters

4.11.3 Rupture of final HEPA filters-extensive vibration and defective equipment

4.11.4 Rupture of HEPA filters - sudden increase in flow resistance

5.1.6 Filter fire

5.1.8 Off-gas momentarily by-passes HEPA filter

4.4.3[a] Major fire in feed preparation line (FRP-ILW line)

4.4.5[a] Incinerator explosion (FRP-ILW)

4.4.4[a] Explosion in feed preparation system

4.5.2 Incinerator flame-out explosion

4.5.3 Solvent fire

4.5.4 Explosion in incinerator

4.7.8 Cell HEPA failure with bitumen fire (in FRP-TRU process line)

4.7.7 Cell HEPA failure with any minor accident

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a. Accident with potential for increased worker exposure

MODERATE ACCIDENTS

4.9.13 Process shutdown with volatile venting - 1 day

- 4.9.3 Process shutdown while dissolver is operating
- 4.10.3 VOG Process shutdown
- 4.10.6 Process shutdown

4.9.14 Process shutdown with carbon-14 and krypton-85 venting - 30 days

- 4.9.8 Process shutdown with ^{14}C venting
- 4.9.11 Process shutdown with ^{85}Kr venting

5.2.1[(a)] Waste Zr hulls canister breached by drop

- 6.4.4 Collision or derailment subjects fuel residues cask to severe impact and fire

5.3.6[(a)] Tornado strikes

- 5.3.5[(a)] Fire in storage stack

5.3.8[(a)] Crane drops drum

5.4.19 Canister failure in receiving cell

- 5.4.20 Canister failure in weld and test cell

5.5.4 Storage container leakage

- 5.5.3 Decontamination trash fire

5.6.4[(a)] Kr cylinder ruptured in operating area or storage corridor

- 4.9.12[(a)] Oxygen recombiner explosion
- 5.6.2 Kr cylinder ruptured in storage cell or hot cell
- 5.6.3 Kr cylinder corrodes

5.7.18 Fuel handling accident

- 5.7.7[(a)] Fuel handling accident
- 5.7.8[(a)] Fuel handling accident without confinement protection
- 5.7.9[(a)] Rail cask venting
- 5.7.11[(a)] Dropped fuel transfer basket

5.7.39[(a)] Packaged element fails in storage

- 5.7.33[(a)] Packaged element fails in storage
- 5.7.36[(a)] Packaged element fails in storage

6.2.6 Collision or derailment causes damage to rail cask mechanical cooling system

- 6.2.5 Fire accompanying accident causes rail cask cavity to overpressurize; relief valve operates

6.2.13 Fire accompanying accident causes truck cask cavity to overpressurize; relief valve operates

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a. Accident with potential for increased worker exposure

MODERATE ACCIDENTS

6.3.4 Collision or derailment results in loss of neutron shielding from solid high-level waste cask

6.2.4 Collision or derailment results in loss of neutron shielding from spent fuel rail cask

6.2.12 Truck collision or overturn results in loss of neutron shielding from spent fuel truck cask

7.6 Waste package dropped down mine shaft

7.5[a] Waste container drop during handling

7.8[a] LLW drum rupture due to mechanical damage and fire

7.9[a] LLW drum rupture due to internal explosion

7.7 Tornado strikes mined materials storage area

SEVERE ACCIDENTS

5.5.5 Storage container breach

5.5.6[a] Plutonium product criticality

5.7.12 Design basis tornado

5.4.13[a] Contamination of secondary cooling water

5.4.15 Design basis tornado

5.4.27 Design basis tornado

5.7.13[a] Criticality

5.7.28[a] Criticality

5.7.20 Dropped fuel assembly - rupture 20% of the rods

6.2.8 Cavity coolant lost from spent fuel rail cask; no emergency action taken

6.3.5 Collision or derailment subjects high-level waste cask to severe impact and fire

6.2.7 Collision or derailment causes spent fuel rail cask to be subjected to severe impact and fire

6.2.14 Collison or overturn causes spent fuel truck cask to be subjected to severe impact and fire

6.6.4 Non-high-level TRU waste container is subjected to severe impact and fire

NON DESIGN BASIS ACCIDENTS

7.11 Repository breach by meteorite

7.12 Repository breach by drilling

7.13 Repository breach by solution mining

7.15 Repository breach by faulting with ground water transport

7.14 Volcanism

7.16 Erosion

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a. Accident with potential for increased worker exposure

APPENDIX 3B

BASES FOR CRITICALITY CALCULATIONS

APPENDIX 3B

BASES FOR CRITICALITY CALCULATIONS

COMPUTER CODES AND NUCLEAR CROSS SECTIONS

The criticality calculations for the various facilities that have significant quantities of fissile material were performed by means of a variety of computer codes. Values of k_∞ and multigroup cross sections were calculated by the code GAMTEC-II[1] from a library based on ENDF/B-IV cross sections. Values of k_{eff} for finite systems were calculated by means of the Monte Carlo criticality code KENO IV[2] and the one-dimensional diffusion theory code HFN.[3] Most of the work was done with KENO IV. HFN was used only for a few of the well-thermalized systems.

HLLW SOLUTION FROM URANIUM-ONLY RECYCLE, WITH PLUTONIUM DISCARDED TO HLW

The criticality calculations for HLLW solution that contains all the discarded plutonium (Case 2a) are based on the following inputs:

- The bulk volume/unit mass measurement is 567 ℓ/MTHM
- The HLLW contents (1185 g/ℓ) are summarized in Table 3.B.1.
- The solution contains 4.777 kg U/MTHM and 8.410 kg Pu/MTHM. Table 3.B.2 lists the isotopic compositions of U and Pu as derived from the ORIGEN run for Case 2a.

TABLE 3.B.1. HLLW Containing Discarded Pu - Case 2a, g/ℓ

H	2.0
Na	0.147
Fe	2.0
Ni	0.133
Cr	0.333
Gd	15.9
Fission products	44.7
Actinides	1.96
U	8.43
Pu	14.83
Activation products	0.520
PO_4	2.67
NO_3	199
H_2O	892

TABLE 3.B.2. Isotopic Composition of U and Pu Contained in HLLW - Case 2a

Isotope	kg/MTHM	wt% of Total U	wt% of Total Pu
^{232}U	2.48×10^{-9}	0	
^{233}U	1.44×10^{-2}	0	
^{234}U	2.87×10^{-5}	0	
^{235}U	3.94×10^{-2}	0.82	
^{236}U	2.51×10^{-2}	0.53	
^{237}U	1.60×10^{-10}	0	
^{238}U	4.71	98.65	
^{236}Pu	6.75×10^{-7}		0
^{238}Pu	1.85×10^{-1}		2.20
^{239}Pu	4.74		56.32
^{240}Pu	2.04		24.26
^{241}Pu	1.04		12.42
^{242}Pu	4.04×10^{-1}		4.80

These inputs, within the limits imposed by the GAMTEC-II computer code and the ENDF/B-IV cross-section library, provide the basis for the nuclides and nuclei densities listed in Table 3.B.3 to represent the Case 2a HLLW solution.

TABLE 3.B.3. Nuclides and Nuclei Densities for the HLLW Solution - Case 2a, atoms/barn-cm

	Atoms/barn-cm
^{235}U	1.79×10^{-7}
^{238}U	2.11×10^{-5}
^{239}Pu	2.12×10^{-5}
^{240}Pu	9.09×10^{-6}
^{241}Pu	4.63×10^{-6}
^{242}Pu	1.78×10^{-6}
Gd	6.10×10^{-5}
N	1.93×10^{-3}
O	3.57×10^{-2}
H	6.09×10^{-2}

Because GAMTEC-II can accept only ten isotopes plus elements, it is necessary to omit the nuclides present in small amounts and those that are judged to have a small effect on the reactivity. Fission products are also omitted because they are not available from the GAMTEC-II library. Because the fission products are generally poor neutron moderators and fair neutron absorbers, this omission introduces a small degree of conservatism; i.e., calculated reactivities are slightly too large.

SPRAY CALCINE FROM URANIUM-ONLY RECYCLE WITH PLUTONIUM DISCARDED TO HLW

The criticality calculations for spray calcine product that contains all the discarded plutonium (Case 2a) are based on the following inputs:

- The bulk volume/unit mass = 43.7 ℓ/MTHM with a void fraction of 70 vol%.
- The spray calcine contents (61.942 kg calcine/MTHM) are summarized in Table 3.B.4.
- The calcine contains 4.777 kg U/MTHM and 8.410 kg Pu/MTHM as derived from the ORIGEN run for Case 2a. (The isotopic compositions are identical with those listed in Table 3.B.1.)

TABLE 3.B.4. Calcine Containing Discarded Pu - Case 2a, kg solid/MTHM

	kg Solid/MTHM
Na_2O	0.112
Fe_2O_3	1.622
NiO	0.096
Cr_2O_3	0.276
Gd_2O_3	10.372
FP_xO_y	31.440
Act_zO_g	1.220
UO_3	5.741
PuO_2	9.533
AP_rO_t	0.400
P_2O_5	1.130

These inputs, within the limits imposed by the GAMTEC-II computer code and the ENDF/B-IV cross-section library, provide the basis for the nuclides and nuclei densities listed in Table 3.B.5 to represent the case 2a spray calcine product.

TABLE 3.B.5. Nuclides and Nuclei Densities for Spray Calcine - Case 2a, atoms/barn-cm

	Atoms/barn-cm
^{235}U	2.32×10^{-6}
^{238}U	2.74×10^{-4}
^{239}Pu	2.75×10^{-4}
^{240}Pu	1.18×10^{-4}
^{241}Pu	6.01×10^{-5}
^{242}Pu	2.31×10^{-5}
O	9.68×10^{-3}
Gd	7.89×10^{-4}

Because GAMTEC-II can accept only ten isotopes and elements, only eight nuclides have been considered for spray calcine so that boron and hydrogen could be included in calculations

involving additional mixtures. Fission products are also omitted. This results in a small degree of conservatism; i.e., calculated reactivities are slightly too large.

In the calculations involving slurry mixtures of the spray calcine product and HLLW solution, the density of the solid calcine is assumed to be constant (at the value corresponding to the dry void volume fraction of 70%) when the volume fraction of solution is less than 70%. After the void space is completely filled, the density of the solid calcine is reduced as more solution is added. Any volume changes due to the dissolution of soluble components in the calcine are ignored.

FLUIDIZED BED CALCINE FROM URANIUM-ONLY RECYCLE WITH PLUTONIUM DISCARDED TO HLW

The criticality calculations for fluidized bed calcine with plutonium discarded to HLW (Case 2a) are based on the following inputs:

- Bulk density = 2.44 g calcine/cm^3 with a void fraction of 20 vol%.
- The fluidized bed calcine contents (61.942 kg calcine/MTHM) are identical to that of spray calcine and are summarized in Table 3.B.4.
- The calcine contains 4.777 kg U/MTHM and 8.410 kg Pu/MTHM as derived from the ORIGEN run for Case 2a. (The isotopic compositions are identical with those listed in Table 3.B.1.)

These inputs, within the limits imposed by the GAMTEC-II computer code and the ENDF/B-IV cross-section library, provide the basis for the nuclides and nuclei densities listed in Table 3.B.6 to represent the dry Case 2a fluidized bed calcine.

TABLE 3.B.6. Nuclides and Nuclei Densities for Fluidized Bed Calcine - Case 2a, atoms/barn-cm

	Atoms/barn-cm
^{235}U	3.99×10^{-6}
^{238}U	4.74×10^{-4}
^{238}Pu	1.85×10^{-5}
^{239}Pu	4.72×10^{-4}
^{240}Pu	2.03×10^{-4}
^{241}Pu	1.03×10^{-4}
^{242}Pu	3.98×10^{-5}
O	1.67×10^{-2}
Gd	1.36×10^{-3}

Because GAMTEC-II can accept only ten isotopes plus elements; Na, Ni, Cr, activation products, P, ^{232}U, ^{233}U, ^{234}U, ^{237}U, and ^{236}Pu effects are considered negligible. The Fe, Np, Am, and Cf are present in small quantities and can be neglected without affecting the calculational results. The ^{236}U is included as ^{238}U. Fission products are 25.197 kg/MTHM or 40.7% of the total calcine by weight; however, they are generally poor neutron moderators and fair neutron absorbers. If their results had been taken into account, it would have reduced the calculated reactivities. All the oxygen in the entire calcine was represented. For the wet systems,

hydrogen in the water was also represented. In summary, 94% of the entire actinides were represented including all plutonium and uranium except trace quantities, and 54% of the total calcine by weight was represented. Had the other 46% by weight also been represented, it would have decreased the calculated reactivities. Therefore, the calculational results presented are all concervative.

For calculations in which the Case 2a fluidized bed calcine was mixed with water, the calcine particles were assumed to occupy a certain fixed volume of space whether wet or dry. Calcine with a bulk density of 2.44 g calcine/cm^3 occupies 80% of all space and in the corresponding wet case the water occupies the remaining 20% of all space. This is modified, for example, to 1.22 g calcine/cm^3 occupying 40% of all space and water occupying the remaining 60% of all space. This assumption is only an approximation because some of the oxides in the calcine go into solution, and the resultant volume is not the same as the sum of the original volumes. This approximation will suffice for the conceptual study. The resulting uncertainty in criticality calculations will be small. Final conclusions are unchanged.

An example of the nuclei densities used in representing 200 g Pu/ℓ and 10 g Gd/ℓ, wet fluidized bed calcine (equivalent to 1.473 g calcine/cm^3) is summarized in Table 3.B.7.

TABLE 3.B.7. Nuclides and Nuclei Densities for Wet Fluidized Bed Calcine, 200 g Pu/ℓ - Case 2a, atoms/barn-cm

	Atoms/barn-cm
H	3.46×10^{-2}
O	2.614×10^{-2}
Gd	3.830×10^{-5}
^{235}U	2.40×10^{-6}
^{238}U	2.85×10^{-5}
^{238}Pu	1.11×10^{-15}
^{239}Pu	2.84×10^{-4}
^{240}Pu	1.22×10^{-4}
^{241}Pu	6.21×10^{-5}
^{242}Pu	2.39×10^{-5}

BOROSILICATE GLASS FROM URANIUM-ONLY RECYCLE, WITH PLUTONIUM DISCARDED TO HLW

The criticality calculations for borosilicate glass containing calcine, with plutonium discarded to HLW (Case 2a) are based on the following inputs:

- Bulk volume/unit mass is 58.3 ℓ/MTHM.
- The borosilicate glass contents (175.017 kg glass/MTHM) are summarized in Table 3.B.8.
- The glass contains 4.777 kg U/MTHM and 8.410 kg Pu/MTHM as derived from the ORIGEN run for Case 2a. (The isotopic compositions are identical with those listed in Table 3.B.1.)

These inputs, within the limits imposed by the GAMTEC-II computer code and the ENDF/B-IV cross-section library, provide the basis for the nuclides and nuclei densities listed in Table 3.B.9 to represent the dry case 2a calcine in borosilicate glass.

TABLE 3.B.8. Borosilicate Glass Containing Discarded Pu - Case 2a, kg solid/MTHM

NaO_2	0.112 kg solid/MTHM from calcine
Fe_2O_3	1.622 kg solid/MTHM from calcine
NiO	0.096 kg solid/MTHM from calcine
Cr_2O_3	0.276 kg solid/MTHM from calcine
Gd_2O_3	10.372 kg solid/MTHM from calcine
FP_xO_y	31.440 kg solid/MTHM from calcine
Act_zO_g	1.220 kg solid/MTHM from calcine
UO_3	5.741 kg solid/MTHM from calcine
PuO_2	9.533 kg solid/MTHM from calcine
AP_rO_t	0.400 kg solid/MTHM from calcine
P_2O_5	1.130 kg solid/MTHM from calcine
SiO_2	66.5 kg solid/MTHM from frit
B_2O_3	18.4 kg solid/MTHM from frit
TiO_2	7.35 kg solid/MTHM from frit
Al_2O_3	1.83 kg solid/MTHM from frit
Na_2O	15.6 kg solid/MTHM from frit
K_2O	9.18 kg solid/MTHM from frit
CaO	3.68 kg solid/MTHM from frit

TABLE 3.B.9. Nuclides and Nuclei Densities for Borosilicate Glass - Case 2a, atoms/barn-cm

	Atoms/barn-cm
^{10}B	1.081×10^{-3}
O	4.507×10^{-2}
Na	5.238×10^{-3}
Si	1.143×10^{-2}
Gd	5.912×10^{-4}
^{238}U (includes ^{236}U)	2.056×10^{-4}
^{239}Pu (includes ^{235}U)	2.065×10^{-4}
^{240}Pu (includes ^{238}Pu)	9.586×10^{-5}
^{241}Pu	4.479×10^{-5}
^{242}Pu	1.725×10^{-5}

The isotopes listed in Table 3.B.9 were chosen because GAMTEC-II can accept only ten isotopes and elements. The elements B, Na, and Si are important constituents from the glass frit, and hydrogen is not present. Of the combined uranium and plutonium isotopes, ^{236}U represents only 0.53% of the ^{238}U + ^{236}U input, ^{235}U represents only 0.84% of the ^{239}Pu + ^{235}U input, and ^{238}Pu is only 8.38% of the ^{240}Pu + ^{238}Pu. Therefore, the ^{236}U and ^{235}U isotopes are essentially negligible anyway. The combination of ^{240}Pu and ^{238}Pu has a smaller effect than does neglecting the sodium in the calculation. All the oxygen in both the calcine and the glass frit was represented. The sodium from both was also represented. In summary, 83.5% of

of the total glass frit by weight was represented, giving a total of 77.5% of the total glass plus calcine by weight. If the other portion of the weight had also been represented, the reactivity of the system would be changed only on the order of a percent.

For calculations in which it was assumed that the PuO_2 could concentrate by some mechanism, only the PuO_2 was assumed to become more dense. This effect was assumed to decrease the density of all other components in the calcine and glass frit by the same fraction. Therefore, both gadolinium and uranium densities were decreased as plutonium increased. Because the PuO_2 is assumed to be finely divided, it was assumed to occupy a volume such that it would just fill all space at theoretical density of 11.46 g PuO_2/cm^3. Some representative nuclei densities are listed in Table 3.B.10.

TABLE 3.B.10. Representative Nuclei Densities for Borosilicate Glass in Which PuO_2 has Become Concentrated, atoms/barn-cm

	Nuclei Density (atoms/barn-cm)		
	5.60 g PuO_2/cm^3, 4.94 g Pu/cm^3	1.40 g PuO_2/cm^3, 1.24 g Pu/cm^3	0.35 g PuO_2/cm^3, 0.309 g Pu/cm^3
^{10}B	5.529×10^{-4}	9.492×10^{-4}	1.048×10^{-3}
O	4.750×10^{-2}	4.513×10^{-2}	4.454×10^{-2}
Na	2.678×10^{-3}	4.598×10^{-3}	5.078×10^{-3}
Si	5.847×10^{-3}	1.004×10^{-2}	1.108×10^{-2}
Gd	3.023×10^{-4}	5.190×10^{-4}	5.732×10^{-4}
^{238}U (includes ^{236}U)	1.051×10^{-4}	1.805×10^{-4}	1.993×10^{-4}
^{239}Pu (includes ^{235}U)	7.011×10^{-3}	1.754×10^{-3}	4.398×10^{-4}
^{240}Pu (includes ^{238}Pu)	3.282×10^{-3}	8.204×10^{-4}	2.051×10^{-4}
^{241}Pu	1.534×10^{-3}	3.834×10^{-4}	9.584×10^{-5}
^{242}Pu	5.905×10^{-4}	1.476×10^{-4}	3.691×10^{-5}

PuO_2 IN THE INTERIM PLUTONIUM STORAGE FACILITY

The criticality calculations for PuO_2 in the interim plutonium storage facility (Case 2b) are based on the following inputs:

- The bulk density of PuO_2 is 3.5 g/cm^3 with a void fraction of 0.6946.
- The H_2O content is 1 wt%.
- The isotopic composition is ^{239}Pu, 95 wt%; ^{240}Pu, 5 wt%.
- Nuclei densities are listed in Table 3.B.11.

TABLE 3.B.11. Nuclides and Nuclei Densities for PuO_2 - Case 2b, atoms/barn-cm

	Atoms/barn-cm
^{239}Pu	7.39×10^{-3}
^{240}Pu	3.89×10^{-4}
H	2.34×10^{-3}
O	1.67×10^{-2}

OTHER MATERIALS USED IN CRITICALITY SAFETY ANALYSIS

The ENDF/B-IV cross sections prepared with the GAMTEC-II computer code were based on the data in Table 3.B.12.

TABLE 3.B.12. Nuclei Densities for Materials Used in Criticality Safety Analysis, atom/barn-cm

Material	Isotope or Element	Nuclei Density, atoms/barn-cm
Water	H	6.68652×10^{-2}
Density 1.000 g/cm^3	O	3.34326×10^{-2}
Concrete (Type 04 from ANL-5800 or ANL-6443)		
Density 2.336 g/cm^3	H	7.768×10^{-3}
	O	4.386×10^{-2}
	Na	1.048×10^{-3}
	Mg	1.486×10^{-4}
	Al	2.389×10^{-3}
	Si	1.580×10^{-2}
	K	6.931×10^{-4}
	Ca	2.915×10^{-3}
	Fe	3.128×10^{-4}
Granite	H	2.16×10^{-3}
Density 2.729 g/cm^3	C	3.74×10^{-5}
	O	4.84×10^{-2}
	Na	2.06×10^{-3}
	Mg	1.44×10^{-3}
	Al	5.00×10^{-3}
	Si	1.64×10^{-2}
	K	1.11×10^{-3}
	Ca	1.51×10^{-3}
	Fe	1.52×10^{-3}
Shale	H	1.01×10^{-2}
Density 2.980 g/cm^3	C	2.29×10^{-3}
	O	5.63×10^{-2}
	Na	7.54×10^{-4}
	Mg	1.09×10^{-3}
	Al	5.46×10^{-3}
	Si	1.75×10^{-2}
	K	1.25×10^{-3}
	Ca	1.00×10^{-3}
	Fe	1.52×10^{-3}

TABLE 3.B.12. (contd)

Material	Isotope or Element	Nuclei Density, atoms/barn-cm
Stainless Steel	18 wt% Cr	1.65136×10^{-2}
Density 7.92 g/cm^3	74 wt% Fe	6.32077×10^{-2}
	8 wt% Ni	6.50004×10^{-3}
Depleted Uranium	0.25 wt% ^{235}U	1.21398×10^{-4}
Density 18.95 g/cm^3	99.75 wt% ^{238}U	4.78261×10^{-2}
Lead	Pb	3.29928×10^{-2}
Density 11.35 g/cm^3		
Graphite	C	9.030×10^{-2}
Density 1.80 g/cm^3		
Rock Salt	Na	2.248×10^{-2}
Density 2.18 g/cm^3	Cl	2.248×10^{-2}
Saturated Brine Solution	H	5.8871×10^{-2}
Density 1.198 g/cm^3	O	2.9435×10^{-2}
	Na	3.2682×10^{-3}
	Cl	3.2682×10^{-3}

ACRONYMS LIST

ACRONYMS LIST

A-E	architect-engineer
AAPG	American Association of Petroleum Geologists
ACVSF	air-cooled vault storage facility
AEC	Atomic Energy Commission
AECL	Atomic Energy of Canada, Limited
AFR	away from reactor (spent fuel storage)
AGNS	Allied General Nuclear Services
ALARA	as low as reasonably achievable
AMAD	aerodynamic median activity diameter
AP	activation product
API	American Petroleum Institute
APS	atmospheric protection system
BFRSS	Barnwell Fuel Receiving and Storage Station
BIF	bitumen immobilization facility
BPPF	Barnwell Plutonium Product Facility
BTU	British thermal unit
BWR	boiling water reactor
CANDU	Canadian heavy water reactor
CDC	canister decontamination cell (cubicle)
CFR	Code of Federal Regulations
CIF	cement immobilization facility
CRWM	Committee on Radioactive Waste Management
CUP	cask unloading pool
CVCS	chemical and volume control system
CW	canistered waste
CWMS	Generic Environmental [illegible]e-ment on Commercial Ra[illegible] Waste Management, DOE-[illegible]
CWTF	cask weld test faci[illegible]
DCSF	dry caisson storage f[illegible]y
DF	decontamination factor
DOE	Department of Energy
DOG	dissolver off-gas
DOP	dioctyphthalate
DOT	Department of Transportation
DTPA	diethylenetriamine pentaacetic acid
ECWS	essential cooling water system
EPC	engineering, procurement, and construction
ER	environmental report
ERDA	Energy Research and Development Administration
ESFS	engineered safety features systems
ESFS	essential spray pond system
FFTF	Fast Flux Test Facility
FP	fission product
FPF	fuel packaging facility
FRP	fuel reprocessing plant
FRPF	fuel residue packaging facility
FRSSF	fuel residue subsurface storage facility
FRVSF	fuel residue vault storage facility
FRW	fuel residue waste
FSA	fuel storage area
FSAR	Final Safety Analysis Report
FSB	fuel storage basin
FTF	fuel transfer facility
FTP	fuel transfer platform
GEIS	Generic Environmental Impact Statement
HCF	hulls compaction facility
HEPA	high-efficiency particulate air (filter)
HEU	highly enriched uranium
HLLW	high-level liquid waste
HLW	high-level waste
HM	heavy metal
HMA	hot maintenance area
HMF	hulls melting facility
HPF	hulls packaging facility
HTD	hulls transfer device
HTGR	high temperature gas-cooled reactor
HVAC	heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
IBC	in-bed combustion
ICPP	Idaho Chemical Processing Plant
IFSF	independent fuel storage facility
IIPSF	independent interim plutonium oxide storage facility

ILLW	intermediate-level liquid waste
ILW	intermediate-level waste
INEL	Idaho National Engineering Laboratory
IPSF	interim plutonium oxide storage facility
ISFS	independent spent fuel storage
ISFSB	independent spent fuel storage basin
ISFSF	independent spent fuel storage facility
LAA	limited access area
LEU	low-enriched uranium
LHD	load-haul-dump
LLW	low-level waste
LN_2	liquid nitrogen
LSA	low specific activity
LWBR	light water breeder reactor
LWR	light water reactor
M&M	men and materials
MFBM	thousand board feet measure
MFRP	General Electric Company's Midwest Fuel Reprocessing Plant
MOX FFP	mixed oxide fuel fabrication plant
MP	mine production
MSRE	molten salt reactor
MTHM	metric ton heavy metal
NAA	normal access area
NAC	Nuclear Assurance Corporation
NAS	National Academy of Sciences
NASA	National Aeronautics and Space Administration
NFS	Nuclear Fuel Services
NHLSW	non-high-level solid waste
NLI	National Lead Industries
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NWTS	National Waste Terminal Storage
ORIGEN	a computer program to calculate isotopic composition of irradiated nuclear fuel
ORNL	Oak Ridge National Laboratory
ONWI	Office of Nuclear Waste Isolation
OWI	Office of Waste Isolation
P-T	partioning and transmutation
PCWS	plant cooling water system
PFRF	packaged fuel receiving facility
PNL	Pacific Northwest Laboratory
POG	process off-gas
PSAR	preliminary safety analysis report
PWR	pressurized water reactor
R&D	research and development
RAA	restricted access area
RBOF	receiving basin for offsite fuel, Savannah River Plant
RCS	reactor coolant system
SCRA	storage cask receiving area
SCSF	surface cask storage facility
SF	spent fuel
SFPF	spent fuel packaging facility
SFRSS	spent fuel receiving and storage station
SFSF	spent fuel storage facility
SHLW	solidified high-level waste
SNM	special nuclear material, i.e., enriched uranium and plutonium
SRP	Savannah River Plant
SSC	sealed storage cask
SSCF	sealed storage cask facility
TBP	tributyl phosphate
TD	theoretical density
TN	Transnuclear Inc.
TRU	transuranic
TSA	transuranic storage area
TWCA	Teledyne Wahchang Albany
U-F	urea-formaldehyde
VE	ventilation exhaust
VOG	vessel off-gas
WBS	water basin storage
WBSF	water basin storage facility
WBSF-PF	water basin storage facility for packaged fuel
WCC	waste calcination cell (cubicle)
WCF	waste calcination facility
WIPP	Waste Isolation Pilot Plant
WTEB	waste tank equipment building
WVC	waste vitrification cell
WVF	waste vitrification facility

MEASUREMENT UNITS AND CONVERSIONS

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MEASUREMENT UNITS AND CONVERSIONS

This report preferentially uses the metric system of measurements as defined by the International System of Units (SI). Common English units are often also included in parentheses. Prefixes used with the metric units are defined as follows:

Prefix	Abbreviation	Factor
giga	G	10^9
mega	M	10^6
kilo	k	10^3
centi	c	10^{-2}
milli	m	10^{-3}
micro	μ	10^{-6}
nano	n	10^{-9}

The following lists identify the symbols used in this report and the factors for converting between the SI and English units.

Symbols for metric units used in this report are:

Symbol	Name
°C[a]	degree Celsius
d[a]	day
g	gram
h (or hr)	hour
ha	hectare
kWh	Kilowatt-hour
J	joule
ℓ	liter
m	meter
min	minute
M	gram-mole/liter
MT	metric ton
MW-hr (or MWh)	megawatt-hour
s (or sec)	second
W	watt

a. Units which are not strictly SI but which are widely used.

Symbols for other units used in this report are:

Symbol	Name
atm	atmospheric pressure
BTU	British thermal unit
Ci	curie
°F	degree Fahrenheit
ft	feet
gal	gallon
in.	inch
lb	pound
MFBM	thousand board feet measure
psi	pounds/square inch
R	roentgen
rem	roentgen equivalent man
yd	yard
yr	year

To convert metric to English, multiply by:

Metric	English	Factor
°C	°F	(°C x 9/5) + 32
cm	inch	0.3937
ha	*acre*	*2.47*
kg	lb	2.205
km	mile	0.6214
ℓ	gal	0.2642
m	ft	3.281
m^2	ft^2	10.76
m^3	MFBM	0.424
m^3	ft^3	35.31
m^3	gal	264.2
m^3	yd^3	1.308
MT	ton	0.9070
W	BTU/hr	3.413
W-s/kg-°C	BTU/lb-°F	2.39×10^{-4}
W/m-°C	BTU/hr-ft-°F	0.576

To convert English to metric, multiply by:

English	Metric	Factor
acre	ha	0.405
BTU	W-hr	0.2931
BTU/lb-°F	W-s/kg-°C	4187
BTU/hr-ft-°F	W/m-°C	1.735
°F	°C	(°F-32) x 5/9
ft	m	0.3048
ft^2	m^2	0.0929
ft^3	m^3	0.0283
gal	ℓ	3.785
gal	m^3	3.785×10^{-3}
inches	cm	2.540
lb	kg	0.4536
mile	km	1.609
MFBM	m^3	2.360
ton	MT	1.103
yd^3	m^3	0.7646